ULNRC-3023

ATTACHMENT FOUR A

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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 ~~of~~ Specification 3.6.1.~~2 and~~ 1.

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

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. The 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.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

REVISION 1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} >200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Δk/k. ~~for four loop operation.~~

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Δk/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 equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Δk/k:

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or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

~~b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;~~

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

~~d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. 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. ~~e. When in MODE 3 or 4,~~ 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.~~

REVISION 2

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% Δk/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 equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); ~~and~~

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

CORE REACTIVITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The measured core reactivity shall be within ±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 4.1.1.1.1.b. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.5.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% $\Delta k/k$ prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1.b, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

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REVISION 1

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3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

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.

delete

REVISION

REACTIVITY CONTROL SYSTEMS

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% $\Delta k/k$ 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 prior to the temperature of one or more of the RCS cold legs exceeding 375°F

delete

TSI 4

REVISION I

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.

delete

CHARGING PUMPS - OPERATING

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% Δk/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 prior to the temperature of one or more of the RCS cold legs exceeding 375°F.

delete

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCE - SHUTDOWN

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 2350 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.

delete

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

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 2350 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% $\Delta k/k$ 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

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

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 2350 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% $\Delta k/k$ 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.

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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:

Any immovability of a control rod initially invokes ACTION Statement 3.1.3.1.a. Subsequently, ACTION Statement 3.1.3.1.a may be exited and ACTION Statement 3.1.3.1.d invoked if either the rod control urgent failure alarm is illuminated or an electrical problem is detected in the rod control system.

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

ACTION a - ~~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.~~ INSERT 1

ACTION b - Be in HOT STANDBY within 6 hours.

ACTION c - POWER OPERATION may continue provided that within 1 hour:

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

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

INSERT 1

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

2. Be in HOT STANDBY within 6 hours.

ACTION (Continued)

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

3. The rod is declared inoperable and the SHUTDOWN MARGIN is greater than or equal to 1.3% $\Delta k/k$. ~~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 d - 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 2

INSERT 2

4.1.3.1.3 Prior to reactor criticality, verify the rod drop time of the individual full-length shutdown and control rods is in accordance with FSAR 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.

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POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

delete

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the 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.

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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, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. INSERT 3A

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 3B

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

INSERT 3A

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

INSERT 3B

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.

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REVISION 1

MOVABLE INCORE DETECTORS

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, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

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 $F^N_{\Delta H}$, $F_Q(Z)$, and F_{xy}.

delete

SEISMIC INSTRUMENTATION

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

| REVISION 2

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE			MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Peak Recording Accelerographs				
a. Radwaste Base Slab	± 1.0 g			1
b. Control Room	± 1.0 g			1
c. ESW Pump Facility	± 1.0 g			1
d. Ctmt Structure	± 2.0 g			1
e. Auxiliary Bldg. SI Pump Suctions	± 1.0 g			1
f. SGB Piping	± 2.0 g			1
g. SGC Support	± 1.0 g			1
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)				
a. Ctmt. Base Slab	± 1.0 g			1
b. Ctmt. Oper. Floor	± 1.0 g			1
c. Reactor Support	± 1.0 g			1
d. Aux. Bldg. Base Slab	± 1.0 g			1
e. Aux. Bldg. Control Room Air Filters	± 1.0 g			1
f. Free Field	± 0.5 g			1
3. Triaxial Response-Spectrum Recorder (Passive)				
a. Ctmt. Base Slab	± 1.0 g			1
4. Triaxial Seismic Switches	ACCELERATION LEVEL			
	North	East	Vertical	
a. OBE Ctmt. Base Slab	0.09 g	0.09 g	0.13 g	1
b. SSE Ctmt. Base Slab	0.13 g	0.14 g	0.20 g	1
c. OBE Ctmt. Oper. Fl.	0.10 g	0.10 g	0.13 g	1
d. SSE Ctmt. Oper. Fl.	0.14 g	0.16 g	0.21 g	1
e. System Trigger	0.01 g	0.01 g	0.01 g	1

delete

REVISION 2

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 Air 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 Annunciators 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.

delete

REVISION 2

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown 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

REVISION 1

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

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

delete

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TSI 14 TSI 40

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:

INSERT 4A

~~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 monitors 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 STANBY 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 monitors 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 4A

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 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 or 18 (Containment Radiation Level or 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 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 monitors less than the Minimum Channels OPERABLE requirement of Table 3.3-10, restore one channel 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.

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 Normal Sump Level	2	1
16. Containment Radiation Level (High Range, GT-RIC-59,-60)	~~N.A.~~ 2	1
17. Thermocouple/Core Cooling Detection System	4/core quadrant	2/core quadrant
18. ~~Unit Vent - High Range Noble Gas Monitor (GT-RIC-21B)~~ Reactor Vessel Level Indicating System	~~N.A.~~ 2	1

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REVISION 1 IF

~~TABLE 3.3-10 (Continued)~~

~~TABLE NOTATIONS~~

~~*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.~~

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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	R ~~N.A.~~ (1)
15. Containment ~~Water~~ Normal Sump Level	M	R
16. Containment Radiation Level (High Range) & GT-RIC-59, -60)	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

TABLE 4.3-7 (Continued)

TABLE NOTATIONS

~~*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.~~

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

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

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REVISION 1

INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.8 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 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

4.3.3.8 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.

delete

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INSTRUMENTATION

WASTE GAS HOLDUP SYSTEM

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 At least one hydrogen and both the inlet and outlet oxygen explosive gas monitoring instrumentation channels for each WASTE GAS HOLDUP SYSTEM recombiner shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.5 are not exceeded.

APPLICABILITY: During WASTE GAS HOLDUP SYSTEM operation.

ACTION:

a. With an outlet oxygen monitor channel inoperable, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours.

b. With both oxygen or both hydrogen channels or both the inlet oxygen and inlet hydrogen monitor channels for one recombiner 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.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of:

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

b. An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 92 days with the use of standard gas samples containing a nominal:

1) One volume percent hydrogen, balance nitrogen and four volume percent hydrogen, balance nitrogen for the hydrogen monitor, and

2) One volume percent oxygen, balance nitrogen, and four volume percent oxygen, balance nitrogen for the inlet oxygen monitor, and

3) 10ppm by volume oxygen, balance nitrogen and 80ppm by volume oxygen, balance nitrogen for the outlet oxygen monitor.

delete
REVISION 1

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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 maintained, calibrated, tested, and inspected in accordance with the Callaway Plant's Turbine Overspeed Protection Reliability Program. Adherence to this program shall demonstrate OPERABILITY of this system. The program and any revisions should be reviewed and approved in accordance with Specification 6.5.1.6o. Revisions shall be made in accordance with the provisions of 10 CFR 50.59.

*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.

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REVISION 1

3/4.4.2 SAFETY VALVES

SHUTDOWN

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.

TSI 3

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 PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) 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 hou . and in HOT SHUTDOWN within the following 6 hours.

c. With both PORV(s) 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 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 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 performance of 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 4B →

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

INSERT 4B

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.

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REVISION 1

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

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:

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REVISION 1

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.

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.

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 48% 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

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REVISION 1

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 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 circum stances 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.

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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

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REVISION 1

3/4.4.7 CHEMISTRY

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.

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REVISION 2

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.

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REVISION 2

TABLE 4.4-3

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.

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LIMITING CONDITION FOR OPERATION

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.

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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

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.*

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*This surveillance need not be performed on the untested reactor vessel head vent path until the first COLD SHUTDOWN to meet the OPERABILITY requirements.

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.

INSERT 5A

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.

INSERT 5B

INSERT 5C

**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.

INSERT 5A

b. With two centrifugal charging pumps OPERABLE, restore one of the centrifugal charging pumps to an inoperable status within 4 hours.

INSERT 5B

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.

INSERT 5C

* 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.

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

*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.

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CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.20% by weight of the containment air per 24 hours at P_a, 48.1 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, 48.1 psig.

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

ACTION:

a. With the overall integrated containment leakage rate exceeding 1.0 L_a, perform the ACTION of Specification 3.6.1.1.

b. With the as left overall integrated containment leakage rate exceeding 0.75 L_a, restore the overall integrated leakage rate to less than 0.75 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a:

1) Restore the combined leakage rate to less than 0.60 L_a within 4 hours, or

2) Isolate each failed penetration within 4 hours by use of at least one closed manual valve or blind flange, or a deactivated automatic valve secured in the closed position, or

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

SURVEILLANCE REQUIREMENTS

4.6.1.2 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:

delete

SURVEILLANCE REQUIREMENTS (Continued)

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 P_a, 48.1 psig, during each 10-year service period.

b. If any periodic as found Type A test fails to meet L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive as found Type A tests fail to meet L_a, a Type A test shall be performed at least every 18 months until two consecutive as found Type A tests meet L_a, at which time the above test schedule may be resumed. The as left overall integrated containment leakage rate shall be less than 0.75 L_a;

* A one-time extension of the test interval is allowed for the third Type A test of the first 10-year service period, provided unit shutdown occurs no later than March 31, 1995 and performance of the Type A test occurs prior to unit restart following Refuel 7.

delete

SURVEILLANCE REQUIREMENTS (Continued)

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,

   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.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 48.1 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks, and

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

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 Specifications 4.6.1.7.2 and 4.6.1.7.4, as applicable; and

g. The provisions of Specification 4.0.2 are not applicable.

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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 1.5, 3.5 and 5.5 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:

delete

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

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).

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 determine 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 $\pm$ 10% of the specified installation pressure,

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

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, 48.1 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.~~ FSAR 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 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.

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CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

a. With one containment hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b. With both hydrogen analyzers inoperable, restore at least one analyzer to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 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.

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REVISION 1

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

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.

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REVISION 1

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 in lieu of 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 of 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 No. 67 .

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c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the 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; and (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 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; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

* The surveillance of the alternate charging line portion of the Chemical and Volume Control System may be extended until prior to startup following the next entry into Mode 3 or November 1, 1993, whichever occurs first.

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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) A 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 value of "N" and "C" (previous day's total plus current day's increments) shall be plotted on 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 that type of snubbers 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 group have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber 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, 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.

g. Service Life Monitoring (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.

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TABLE 4.7-2

SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS Column A Extend Interval (Notes 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 population or category size 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, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or 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 B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

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TABLE 4.7-2 (continued)

SNUBBER VISUAL INSPECTION INTERVAL (continued)

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 Columns B and C.

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

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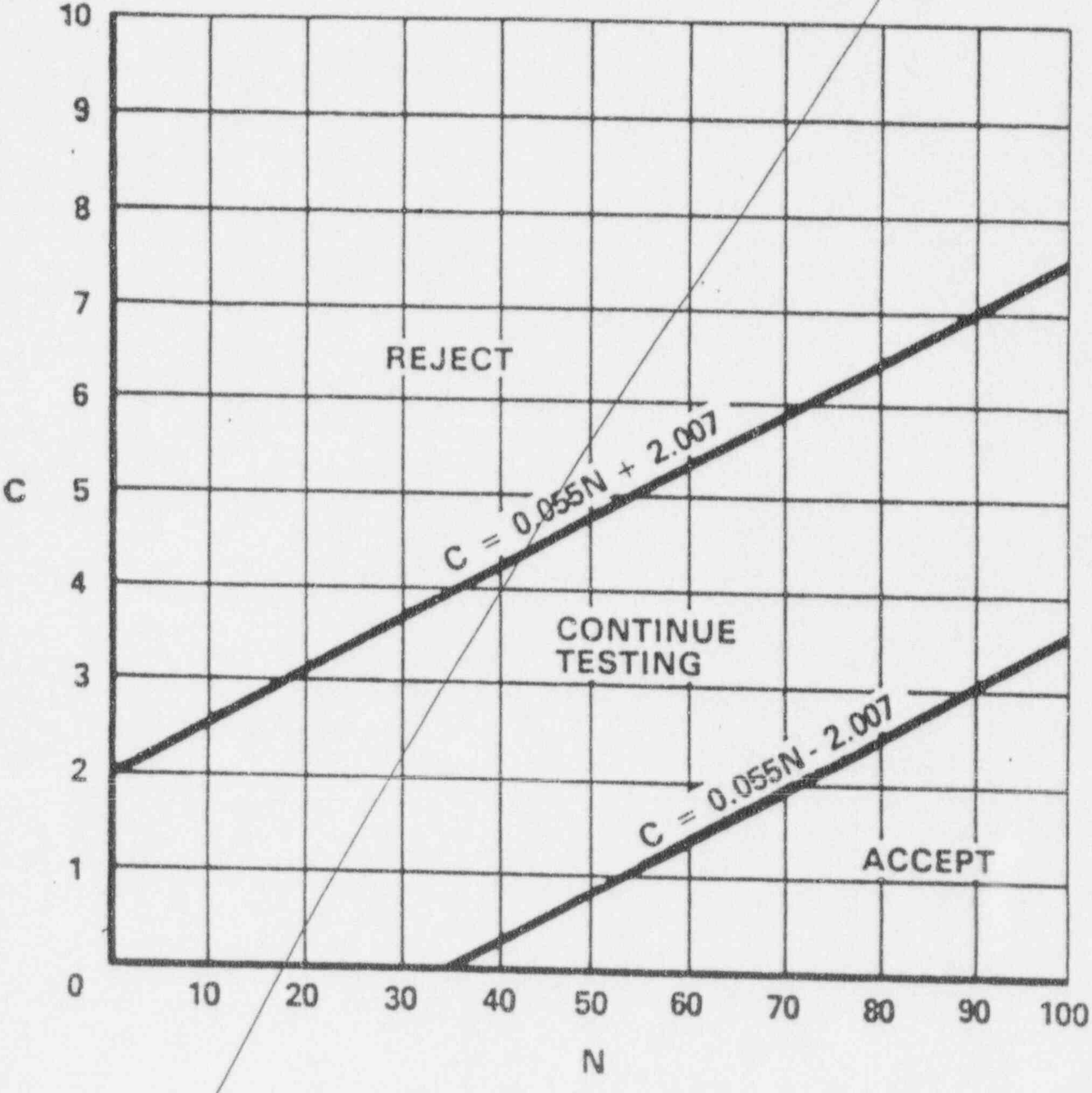

FIGURE 4.7-1

SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

REVISION 1

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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.

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REVISION 1

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.

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.

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3/4.7.12 AREA TEMPERATURE MONITORING

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 (25°F for Electrical Penetration Rooms A and B).

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 (25°F for Electrical Penetration Rooms A and B), 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.

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TABLE 3.7-4

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	106
17.	Electrical Penetration Room B	106
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

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ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices whose circuit limiting fault current exceeds the penetration rating 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 All containment penetration conductor overcurrent protective devices whose circuit limiting fault current exceeds the penetration rating 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

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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.

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3/4.9.5 COMMUNICATIONS

REVISION 1

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.

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REVISION 1

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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 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

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REVISION 1

SURVEILLANCE REQUIREMENTS (Continued)

to removal of the reactor vessel head 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 removal of the reactor vessel head by performing a load test of at least 1250 pounds.

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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, except for the spent fuel pool transfer gates which may be moved over fuel assemblies in the spent fuel pool for refueling activities, fuel handling system maintenance, and transfer gate seal replacement.

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.

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REVISION 1

WATER LEVEL - REACTOR VESSEL

CONTROL RODS

LIMITING CONDITION FOR OPERATION

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.

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REVISION 1

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.

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REVISION 1

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.

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RADIOACTIVE 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 the liner being used to solidify radioactive waste.

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 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.

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RADIOACTIVE 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% 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 on oxygen to less than or equal to 4% by volume, then take ACTION a. above.

c. The provision 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 Specification 3.3.3.10.

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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 Semiannual 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.

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 End of Life (EOL) MTC value. The 300 ppm surveillance limit MTC value 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.

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

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

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

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~~3/4.1.2 BORATION SYSTEMS~~

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The Boration Systems ensure 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 83,745 gallons of 2350 ppm borated water from the RWST. With the RCS average temperature less than 350°F, only one boron injection flow path is required.

3/4.1.1.5 CORE REACTIVITY

When measured core reactivity is within ± 1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the 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 limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the 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. Every accident evaluation is, therefore, dependent upon accurate evaluation of core reactivity. In particular, 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 for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculation models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, 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

INSERT 6 (page 1 of 2)

3/4.1.1.5 CORE REACTIVITY

When measured core reactivity is within $\pm$ 1% $\Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the 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 limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

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

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation is, therefore, dependent upon accurate evaluation of core reactivity. In particular, 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 for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculation models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, 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 BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value shall be 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 BOC 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 their consistency with input to design calculations. Measured core and process parameters 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 restrictions 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 reactor operation.

~~BORATION SYSTEMS (Continued)~~

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~~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% $\Delta k/k$ 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 14,076 gallons of 2350 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.

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.~~

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120 and 228 steps withdrawn for the Control Banks and 18, 210 and 228 steps withdrawn for the Shutdown Banks provides assurance 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. Shutdown and control rods are positioned at 225 steps or higher for fully withdrawn.

MOVABLE CONTROL ASSEMBLIES (Continued)

For purposes of determining compliance with Specification 3.1.3.1, any immovability of a control rod initially invokes ACTION statement 3.1.3.1.a. Subsequently, ACTION statement 3.1.3.1.a may be exited and ACTION statement 3.1.3.1.d invoked if either the rod control urgent failure alarm is illuminated or an electrical problem is 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.

Engineered Safety Features Actuation System Interlocks

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

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

Reactor not tripped - prevents manual block of Safety Injection.

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

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 acutation of Emergency Exhaust or Control Room Emergency Ventilation Systems.

~~3/4.3.3.2 MOVABLE INCORE DETECTORS~~

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~~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 $F_Q(Z)$ or $F^N_{\Delta H}$ 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 ensures that sufficient meteorological data are 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 shutdown 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.

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~~3/4.3.3.8 LOOSE-PART DETECTION SYSTEM~~

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~~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.~~

~~3/4.3.4 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.~~

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BASES

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 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 Vessel Code.

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 and prevent a high pressurizer pressure reactor trip 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 operation is taken in the FSAR analyses for MODE 1, 2 and 3 transients where operation of the PORVs has a beneficial impact on the results of the analysis, 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 operation action.

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~~3/4.4.5 STEAM GENERATORS~~

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~~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) a seismic occurrence greater than the Operating Basis Earthquake; and (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 48% of the tube nominal wall thickness. Steam generator~~

REVISION 1

~~STEAM GENERATORS (Continued)~~

DELETED

~~tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Results from WCAP-10043 have been used to establish plugging limit.~~

~~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 conditions.

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

The CONTROLLED LEAKAGE 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.

OPERATIONAL LEAKAGE (Continued)

The 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 these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which would 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 an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Callaway site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

PRESSURE/TEMPERATURE LIMITS (Continued)

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

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

~~4. The pressurizer heatup and cooldown rates shall not exceed 100°F/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 shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

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

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 17 effective full power years (EFPY) of service life. The 17 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 phosphorus 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 either Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. 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 17 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The lead factor represents the

HEATUP (Continued)

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.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

~~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, two RHR suction relief valves, one RHR suction relief valve and one PORV, 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

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for 1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; 2) a 50°F heat transport effect made

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~~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 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).~~

~~Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 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.~~

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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 7

~~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 accident 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, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.~~

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

~~The following exemptions have been granted to the requirements of Appendix J of 10 CFR Part 50:~~

1. ~~Section III.A.1(a) - an exemption to the requirement to stop the Type A test if excessive leakage is determined. This exemption allows the satisfactory completion of the Type A test if the leakage can be isolated and appropriately factored into the results.~~

2. ~~Section III.A.5(b) - an exemption for the acceptance criteria, in lieu of the present single criterion of the total measured containment leakage rate being less than 0.75 of the maximum allowable leakage rate, L_a, the "as found" allowable leakage rate will be L_a and the "as left" allowable leakage rate will be less than 0.75 L_a.~~

3. ~~Section III.D.1(a) - an exemption that removes the requirement that the third test of each set of three Type A tests be conducted when the plant is shutdown for the 10-year plant inservice inspection.~~

~~*A one-time extension of the test interval is allowed for the third Type A test of the first 10-year service period, as required by Surveillance Requirement 4.6.1.2.a and by Section III.D.1.(a) of Appendix J to 10 CFR Part 50, provided unit shutdown occurs no later than March 31, 1995 and performance of the Type A test occurs prior to unit restart following Refuel 7.~~

INSERT 7

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, 48.1 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, 48.1 psig.

~~CONTAINMENT LEAKAGE (Continued)~~

DELETED

~~Exemption 1 allows the continuance of a Type A test when excessive leakage is~~ found provided that significant leaks are identified and isolated. After completion of the modified Type A test (i.e., a Type A test with the significant leakage paths isolated during the test), local leakage rates of those paths isolated during the modified Type A test will be measured before and after repairs to those paths. The adjusted "as found" leakage rate for the Type A test can be determined by adding the local leakage rates measured, before any repairs to those previously isolated leakage paths, to the containment integrated leakage determined in the modified Type A test plus any leakage improvements (defined below) made prior to the test. This adjusted "as found" leakage rate is to be used in determining the scheduling of the periodic Type A test in accordance with Section III.A.6 of Appendix J.

The acceptability of the modified Type A test can be determined by calculating the adjusted "as left" containment overall integrated leakage rate and comparing it to the acceptance criteria of 0.75 L_a. The adjusted "as left" Type A leakage rate is determined by adding the local leakage rates measured, after any repairs and/or adjustments to those previously isolated leakage paths, to the leakage rate determined in the modified Type A test. It should be noted that additional adjustments for non-standard lineup and changes in containment volume are added to the measured leakage rate for both "as found" and "as left" determinations.

Leakage improvements are defined as the difference between the pre-repair LLRT and post-repair LLRT done on containment penetrations prior to the start of the Type A test.

The only differences between this approach and Appendix J requirements are that: (1) the potentially excessive leakage paths will be repaired and/or adjusted after the Type A test is completed; and (2) the Type A test leakage ~~rate is partially determined by calculation rather than by direct measurement.~~

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.

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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 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 49.5 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~~

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~~This limitation ensures that the structural integrity of the containment~~ vessel 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 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 abnormalties shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances 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 purge 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 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 isolation allowable leakage rate for containment purge supply and exhaust ~~supply~~ valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit ~~of Specification 3.6.1.2b.~~ shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

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~~ 8.4 and 11.0 for the

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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 eductor 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 Containment Cooling System and the Containment Spray System are redundant to each other in providing post-accident cooling of the Containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time 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 GDC 54 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 (or the Purge System) is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The Hydrogen Purge Subsystem discharges directly to the Emergency Exhaust System. Operation of the Emergency Exhaust System with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978.

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 blow down 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 line isolation valves within the closure times of the Surveillance Requirements ~~are~~ is consistent with the assumptions used in the safety analyses.

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES

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~~The OPERABILITY of the feedwater isolation valves: (1) provides a pressure boundary to permit auxiliary feedwater addition in the event of a main steam or feedwater line break inside containment; and (2) ensure that no more than one steam generator will blow down in the event of a steam line rupture which a) minimizes the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and b) limits the pressure rise within containment. The OPERABILITY of the feedwater isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analysis.~~

3/4.7.1.7 STEAM GENERATOR ATMOSPHERIC STEAM DUMP VALVES

The OPERABILITY of the steam generator atmospheric steam dump valves (ASD's) ensures that the reactor decay heat can be dissipated to the atmosphere in the event of a steam generator tube rupture and loss of off-site power and that the Reactor Coolant System can be cooled down for Residual Heat Removal System operation. The number of required ASD'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 ASD is required. In this case, with three ASD's OPERABLE, if the single failure of one ASD occurs and another ASD is assumed to be associated with the faulted steam generator, one ASD remains available for required heat removal.

Each ASD is equipped with a manual block valve (in the auxiliary building) to provide a positive shutoff capability should an ASD develop leakage. Closure of the block valves of all ASD's because of excessive seat leakage does not endanger the reactor core; consistent with plant

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3/4.7.1.6 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 the 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.

3/4.7.1.7 STEAM GENERATOR ATMOSPHERIC STEAM DUMP VALVES (Continued)

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 ASD can be used to control release of steam to the atmosphere. For the steam generator tube rupture event, primary to secondary leakage can be terminated by depressurizing the Reactor Coolant System with the pressurizer power operated relief valves.

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

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~~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.~~

3/4.7.3 COMPONENT COOLING WATER SYSTEM

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

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

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

3/4.7.5 ULTIMATE HEAT SINK

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

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~~3/4.7.8 SNUBBERS~~

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~~All snubbers are required OPERABLE to ensure that the structural integrity~~ of the Reactor Coolant System and all other safety-related systems are 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 On-Site 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 on the number of unacceptable snubbers found during the previous inspection in proportion to the size of the various snubber populations or categories. The Snubber Visual Inspection Interval is determined in accordance with Table 4.7-2. The maximum inspection interval can be as long as two refuel cycles but not more than 48 months, provided the requirements of Table 4.7-2 are met. A snubber is considered unacceptable if it fails the acceptance criteria of visual inspection.

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~~

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~~SNUBBERS (Continued)~~

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~~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 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 limitation 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.~~

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~~SEALED SOURCE CONTAMINATION (Continued)~~

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~~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.~~

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~~3/4.7.12 AREA TEMPERATURE MONITORING~~

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~~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, except for Electrical Penetration Rooms A and B. These rooms have an alarm at ≤ 103°F with a maximum room temperature of 106°F.~~

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~~3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES~~

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~~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, shall be 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 shall be made in accordance with Section 50.59 of 10 CFR Part 50.

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3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

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

3/4.9.2 INSTRUMENTATION

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

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the fuel handling accident radiological consequence and spent fuel pool thermal-hydraulic analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

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

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

The restriction on the setpoint for GT-RE-22 and GT-RE-33 is based on a fuel handling accident inside the Containment Building with resulting damage to one fuel rod and subsequent release of 0.1% of the noble gas gap activity, except for 0.3% of the Kr-85 gap activity. The setpoint concentration of 5E-3 uCi/cc is equivalent to approximately 150 mR/hr submersion dose rate.

~~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.~~

~~The spent fuel pool transfer gates are excluded from this restriction because with a limited gate lift height, the spent fuel pool racks will absorb the impact of a dropped gate without damage to fuel assemblies. In addition, redundant trolleys and supports are used when moving the gates to preclude dropping a gate on the spent fuel racks, the time and distance the gates are moved over fuel is minimized as much as practical, and gate travel over fuel assemblies containing RCCAs is prohibited. The spent fuel pool transfer gates are only moved for refueling activities, fuel handling system maintenance, and to change gate seals.~~

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 140F 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 requirement to maintain a 1000 gpm flowrate ensures that there is adequate flow to prevent boron stratification. The RHR flow to the RCS will provide adequate cooling to prevent exceeding 140F and to allow flowrates which provide additional margin against vortexing at the RHR pump suction while in partial drain operation.

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 residual heat

REVISION 1

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~~

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

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~~3/4.11.1.4 LIQUID HOLDUP TANKS~~

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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.

~~3/4.11.2.5 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.

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SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Senior Vice President-Nuclear within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33;

c. Plant Security Plan implementation;

d. Radiological Emergency Response Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

g. Quality Assurance Program implementation for effluent and environmental monitoring; and

~~h. Turbine Overspeed Protection Reliability Program; and~~

h. ~~i.~~ Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and changes thereto, including temporary changes shall be reviewed prior to implementation as set forth in Specification 6.5 above.

6.8.3 The plant Administrative Procedures and changes thereto shall be reviewed in accordance with Specification 6.5.1.6 and approved in accordance with Specification 6.5.3.1. The associated implementing procedures and changes thereto shall be reviewed and approved in accordance with Specification 6.5.3.1.

6.8.4 The following programs shall be established, implemented and maintained:

a. Reactor Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the Containment Spray System, Safety Injection System, Chemical and Volume Control System, and RHR System. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

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PROCEDURES AND PROGRAMS (Continued)

~~e. Turbine Overspeed Protection Reliability Program~~

~~A program to increase the assurance that the Turbine Overspeed Protection System functions, if challenged, and to assure structural integrity of turbine components which could result in missile generation in the event of an actual overspeed occurrence. The program shall include the following:~~

~~1) Periodic testing and inspection requirements,~~

~~2) Specification of test and inspection intervals, and~~

~~3) Administrative restrictions and procedural guidance for program implementation such as: record keeping; reporting; evaluation and disposition of discrepancies; review and approval of revisions to the program; and authorization(s) required to deviate from the program guidelines.~~

e. ~~f.~~ Radioactive Effluent Controls Program

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

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

PROCEDURES AND PROGRAMS (Continued)

e. ~~f.~~ Radioactive Effluent Controls Program (Continued)

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 50, Appendix B, Table II, Column 1,

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. ~~g.~~ 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 that modifications to the monitoring program are made if required by the results of this census, and

PROCEDURES AND PROGRAMS (Continued)

f. ~~9.~~ Radiological Environmental Monitoring Program (Continued)

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.8.5 INSERT 9

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.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the License involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

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The following programs, relocated from the Technical Specifications to FSAR 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.

RECORD RETENTION (Continued)

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the ORC and the NSRB;

l. Records of the service lives of all hydraulic and mechanical snubbers ~~required by Specification 3.7.8~~ including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality;

n. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed; and

o. Records of reviews performed for changes made to APA-ZZ-01003, the OFFSITE DOSE CALCULATION MANUAL and APA-ZZ-01011, the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to Paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area, or

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CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. All equipment hatches are closed and sealed.

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3.

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

e. The containment leakage rates are within the limits listed in the Bases of Specification 3.6.1.1.

CONTROLLED LEAKAGE

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

CORE ALTERATION

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

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit specific document that provides core operating limits for the current operating reload cycle. The 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.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} \geq 200°F$

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 3 and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.3% $\Delta k/k$, within 15 minutes initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% $\Delta k/k$:

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

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

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SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% $\Delta k/k$, within 15 minutes initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% $\Delta k/k$:

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 SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

CORE REACTIVITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

a. reevaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation, and

b. establish appropriate administrative operating restrictions and Surveillance Requirements, or

c. be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.5.1 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm 1\%$ $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.b. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.5.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% $\Delta k/k$ prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1.b, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

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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:

Any immovability of a control rod initially invokes ACTION statement 3.1.3.1.a. Subsequently, ACTION Statement 3.1.3.1.a may be exited and ACTION Statement 3.1.3.1.d invoked if either the rod control urgent failure alarm is illuminated or an electrical problem is detected in the rod control system.

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

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

2. Be in HOT STANDBY within 6 hours.

ACTION b - Be in HOT STANDBY within 6 hours.

ACTION c - POWER OPERATION may continue provided that within 1 hour:

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

* See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

3. The rod is declared inoperable and the SHUTDOWN MARGIN is greater than or equal to 1.3% $\Delta k/k$. POWER OPERATION may then continue provided that:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.1.3.1.3 Prior to reactor criticality, verify the rod drop time of the individual full-length shutdown and control rods is in accordance with FSAR 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.

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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, except for surveillance testing pursuant to Specification 4.1.3.1.2:

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.6.1 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

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

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* See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

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ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 30 days or prepare and submit a Special Report to the Commission pursuant to 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 or 18 (Containment Radiation Level or 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 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 monitors less than the Minimum Channels OPERABLE requirement of Table 3.3-10, restore one channel 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

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

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Containment Pressure - Normal Range	2	1
2.	Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3.	Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4.	Reactor Coolant Pressure - Wide Range	2	1
5.	Pressurizer Water Level	2	1
6.	Steam Line Pressure	2/steam generator	1/steam generator
7.	Steam Generator Water Level - Narrow Range	2/steam generator	1/steam generator
8.	Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9.	Refueling Water Storage Tank Water Level	2	1
10.	Containment Hydrogen Concentration Level	2	1
11.	Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
12.	Deleted		
13.	Deleted		
14.	Neutron Flux	2	1
15.	Containment Normal Sump Level	2	1
16.	Containment Radiation Level (High Range, GT-RIC-59, -60)	2	1
17.	Thermocouple/Core Cooling Detection System	4/core quadrant	2/core quadrant
18.	Reactor Vessel Level Indicating System	2	1

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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.	Deleted		
13.	Deleted		
14.	Neutron Flux	M	R(1)
15.	Containment Normal Sump Level	M	R
16.	Containment Radiation Level (High Range, GT-RIC-59, -60)	M	R(2)
17.	Thermocouple/Core Cooling Detection System	M	R
18.	Reactor Vessel Level Indicating System	M	R

TABLE 4.3-7 (Continued)

TABLE NOTATIONS

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

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

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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 PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) 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 PORV(s) 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 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 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 performance of 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.

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

SURVEILLANCE REQUIREMENTS (Continued)

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.

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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 and MODE 6 with the reactor vessel head on.*

ACTION:

a. With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

b. With two centrifugal charging pumps OPERABLE, restore one of the centrifugal charging pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

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

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

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* 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 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. 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, 48.1 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to FSAR Section 16.6.1.1 for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

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* 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.

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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, 48.1 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 FSAR 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 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.

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* 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.

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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 End of Life (EOL) MTC value. The 300 ppm surveillance limit MTC value 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.

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

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

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

3/4.1.1.5 CORE REACTIVITY

When measured core reactivity is within ±1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the 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 limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the 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. Every accident evaluation is, therefore, dependent upon accurate evaluation of core reactivity. In particular, 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.

CORE REACTIVITY (Continued)

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the 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 cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, 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 BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value shall be 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 BOC 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 their consistency with input to design calculations. Measured core and process parameters 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 restrictions 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 reactor operation.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120 and 228 steps withdrawn for the Control Banks and 18, 210 and 228 steps withdrawn for the Shutdown Banks provides assurance 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. Shutdown and control rods are positioned at 225 steps or higher for fully withdrawn.

For purposes of determining compliance with Specification 3.1.3.1, any immovability of a control rod initially invokes ACTION statement 3.1.3.1.a. Subsequently, ACTION statement 3.1.3.1.a may be exited and ACTION statement 3.1.3.1.d invoked if either the rod control urgent failure alarm is illuminated or an electrical problem is detected in the rod control system. The rod is considered trippable if the rod was demonstrated OPERABLE during the last performance of Surveillance Requirement 4.1.3.1.2 and met the rod drop time criteria during the last performance of Surveillance Requirement 4.1.3.1.3.

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.

Engineered Safety Features Actuation System Interlocks

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

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

Reactor not tripped - prevents manual block of Safety Injection.

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

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.

3/4.3.3.5 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 shutdown 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.

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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.

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 Vessel Code.

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 and prevent a high pressurizer pressure reactor trip 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 operation is taken in the FSAR analyses for MODE 1, 2 and 3 transients where operation of the PORVs has a beneficial impact on the results of the analysis, 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 operation action.

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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 conditions.

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

The CONTROLLED LEAKAGE 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.

OPERATIONAL LEAKAGE (Continued)

The 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 these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which would 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.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Callaway site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

PRESSURE/TEMPERATURE LIMITS (Continued)

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

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

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

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 17 effective full power years (EFPY) of service life. The 17 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 phosphorus 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 either Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. 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 17 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The lead factor represents the

HEATUP (Continued)

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.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

The OPERABILITY of two PORVs, two RHR suction relief valves, one RHR suction relief valve and one PORV, 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

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for 1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; 2) a 50°F heat transport effect made

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3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

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

Containment leakage rates shall be within the following limits:

1) An overall integrated leakage rate of less than or equal to L_a, 0.20% by weight of the containment air per 24 hours at P_a, 48.1 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, 48.1 psig.

3/4.6.1.3 CONTAINMENT AIR LOCKS

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

3/4.6.1.4 INTERNAL PRESSURE

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

The maximum peak pressure expected to be obtained from a steam line break event is 48 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 49.5 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. Measurement shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

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 purge 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 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 isolation valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

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 8.4 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 eductor 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 Containment Cooling System and the Containment Spray System are redundant to each other in providing post-accident cooling of the Containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time 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 GDC 54 thru 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the Purge System) is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The Hydrogen Purge Subsystem discharges directly to the Emergency Exhaust System. Operation of the Emergency Exhaust System with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978.

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 blow down 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 line isolation valves within the closure times of the Surveillance Requirements is consistent with the assumptions used in the safety analyses.

3/4.7.1.6 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 the 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 analyses.

3/4.7.1.7 STEAM GENERATOR ATMOSPHERIC STEAM DUMP VALVES

The OPERABILITY of the steam generator atmospheric steam dump valves (ASD's) ensures that the 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 ASD'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 ASD is required. In this case, with three ASD's OPERABLE, if the single failure of one ASD occurs and another ASD is assumed to be associated with the faulted steam generator, one ASD remains available for required heat removal.

Each ASD is equipped with a manual block valve (in the auxiliary building) to provide a positive shutoff capability should an ASD develop leakage. Closure of the block valves of all ASD's because of excessive seat leakage does not endanger the reactor core; consistent with plant

3/4.7.1.7 STEAM GENERATOR ATMOSPHERIC STEAM DUMP VALVES (Continued)

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 ASD can be used to control release of steam to the atmosphere. For the steam generator tube rupture event, primary to secondary leakage can be terminated by depressurizing the Reactor Coolant System with the pressurizer power operated relief valves.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

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

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

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

3/4.7.5 ULTIMATE HEAT SINK

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

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3/4.9.1 BORON CONCENTRATION

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

3/4.9.2 INSTRUMENTATION

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

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the fuel handling accident radiological consequence and spent fuel pool thermal-hydraulic analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

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

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

The restriction on the setpoint for GT-RE-22 and GT-RE-33 is based on a fuel handling accident inside the Containment Building with resulting damage to one fuel rod and subsequent release of 0.1% of the noble gas gap activity, except for 0.3% of the Kr-85 gap activity. The setpoint concentration of 5E-3 μCi/cc is equivalent to approximately 150 mR/hr submersion dose rate.

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 requirement to maintain a 1000 gpm flowrate ensures that there is adequate flow to prevent boron stratification. The RHR flow to the RCS will provide adequate cooling to prevent exceeding 140°F and to allow flowrates which provide additional margin against vortexing at the RHR pump suction while in partial drain operation.

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 residual heat

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.

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SAFETY LIMIT VIOLATION (Continued)

c The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Senior Vice President-Nuclear within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33;

c. Plant Security Plan implementation;

d. Radiological Emergency Response Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

g. Quality Assurance Program implementation for effluent and environmental monitoring; and

h. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and changes thereto, including temporary changes shall be reviewed prior to implementation as set forth in Specification 6.5 above.

6.8.3 The plant Administrative Procedures and changes thereto shall be reviewed in accordance with Specification 6.5.1.6 and approved in accordance with Specification 6.5.3.1. The associated implementing procedures and changes thereto shall be reviewed and approved in accordance with Specification 6.5.3.1.

6.8.4 The following programs shall be established, implemented and maintained:

a. Reactor Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the Containment Spray System, Safety Injection System, Chemical and Volume Control System, and RHR System. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

PROCEDURES AND PROGRAMS (Continued)

e. Radioactive Effluent Controls Program

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

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

PROCEDURES AND PROGRAMS (Continued)

e. Radioactive Effluent Controls Program (Continued)

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 50, Appendix B, Table II, Column 1,

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 that modifications to the monitoring program are made if required by the results of this census, and

PROCEDURES AND PROGRAMS (Continued)

f. Radiological Environmental Monitoring Program (Continued)

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.8.5 The following programs, relocated from the Technical Specifications to FSAR 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.

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.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the License involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

RECORD RETENTION (Continued)

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the ORC and the NSRB;

l. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality;

n. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed; and

o. Records of reviews performed for changes made to APA-ZZ-01003, the OFFSITE DOSE CALCULATION MANUAL and APA-ZZ-01011, the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to Paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area, or

ATTACHMENT FIVE

APPLICATION OF POLICY STATEMENT CRITERIA

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 Callaway 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 Wolf Creek 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. Callaway Plant Technical Specifications and Bases (NUREG-1058) 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 Accidents During Shutdown and Refueling."

8. TU Electric letter to NRC, TXX-93098 dated 4-30-93, and NRC Approval and SER dated 11-3-93.

TABLE 1					
Summary of Criteria Application Results Reactivity Control Systems					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	Callaway Results	Note
3.1.1.1	3.1.1.1	Shutdown Margin >200°F	Retain	See Note 1	1
3.1.1.2	3.1.1.2	Shutdown Margin ≤200°F	Retain	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					
Summary of Criteria Application Results Power Distribution Limits					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	Callaway 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					
Summary of Criteria Application Results Instrumentation					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	Callaway 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
3.3.3.8	3.3.3.9	Loose Parts Detection System	Relocate	Relocate	
3.3.3.10		Explosive Gas Monitoring Instrumentation	Not Reviewed	Relocate	6
3.3.4	3.3.4	Turbine Overspeed Protection	Relocate	Relocate	7

TABLE 1					
Summary of Criteria Application Results Reactor Coolant System					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	Callaway 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					
Summary of Criteria Application Results Emergency Core Cooling Systems					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	Callaway 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					
Summary of Criteria Application Results Containment Systems					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	Callaway 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	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					
Summary of Criteria Application Results Plant Systems					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	Callaway 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		Main Feedwater Isolation Valves	Not Reviewed	Retain	
3.7.1.7		Steam Generator Atmospheric Steam Dump Valves	Not Reviewed	Retain	
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	16
3.7.9	3.7.10	Sealed Source Contamination	Relocate	Relocate	
3.7.12	3.7.13	Area Temperature Monitoring	Relocate	Relocate	17

TABLE 1					
Summary of Criteria Application Results Electrical Power Systems					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	Callaway 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					
Summary of Criteria Application Results Refueling Operations					
Tech Spec Number	STS Rev 5 Number	Technical Specification Title	NRC Results	Callaway 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	18
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					
Summary of Criteria Application Results Special Test Exceptions					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	Callaway Results	Note
3.10.1	3.10.1	Shutdown Margin	Relocate	Delete	19
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	19

TABLE 1					
Summary of Criteria Application Results Radioactive Effluents					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	Callaway Results	Note
3.11.1.4	3.11.1.4	Liquid Holdup Tanks	Relocate	Relocate	20
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	20

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, Union Electric 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 3.1.3.1.a and 3.1.3.1.c 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 4.1.1.1.1.a for Modes 1 and 2 has been incorporated into Actions 3.1.3.1.a and 3.1.3.1.c of LCO 3.1.3.1. SR 4.1.1.1.1.b 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 4.1.1.1.1.c, 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 surveillances for inoperable rods and insertion limits with their appropriate LCOs.

   SR 4.1.1.1.1.d 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.

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 RV 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 FSAR Section 16.1.3.2. This is consistent with STS.

5. The Regulatory Guide 1.97, Rev. 2, Type A variables identified in FSAR 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 1.

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 loss of the decay heat removal function in Mode 5 and Mode 6 with vessel head installed by allowing SI 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 4.6.1.1.c will be modified to eliminate reference to a specification that was relocated and instead reference corresponding FSAR 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 FSAR 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. This specification will be relocated and a Snubber Inspection Program statement will be included in new Section 6.8.5.

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

18. 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. Therefore, this LCO will be relocated.

19. 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 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 Specification 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.

20. This specification will be relocated and a Storage Tank Radioactivity Monitoring Program statement will be included in new 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 SDM 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
3.3.3.8	LOOSE-PART MONITORING INSTRUMENTATION
3.3.3.10	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 and 4. For Modes 1 and 2, the criterion is not satisfied.

YES	NO		
___	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 LCO 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 an 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 and 4, 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, does 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; this TS, therefore, does not satisfy criterion 3.

Further, Ref. 4 concluded that the LCO could be relocated for Modes 1 and 2 but must be retained for Modes 3 and 4. However, Union Electric 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 3.1.3.1.a and 3.1.3.1.c 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 4.1.1.1.1.a for Modes 1 and 2 has been incorporated into Actions 3.1.3.1.a and 3.1.3.1.c of LCO 3.1.3.1. SR 4.1.1.1.1.b 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 4.1.1.1.1.c, 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 surveillances for inoperable rods and insertion limits with their appropriate LCOs.

SR 4.1.1.1.1.d 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.

Parameters with design limits such as SDM, MTC, rod drop time, AFD, F_Q, $F\Delta H$, quadrant power tilt ratio, DNBR, pressurizer and SG pressure and temperature limits are chosen to preclude events from occurring that are non-mechanistically examined in FSAR Chapters 6 and 15. These parameters are not modelled in the PSA which is a best-estimate study of plant design vulnerabilities. As such, though these limits are significant for criterion 2 in Modes 3 and 4, they are not significant for 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 and 4 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 LCO 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 which is desirable, yet 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, 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 Callaway IPE does not address initiating events while operating in the shutdown modes. However, based on Refs. 6-8, this TS has not been identified as a significant risk contributor.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Table 3.1.3-2 of the Callaway IPE notes that boration is not required if hot shutdown conditions are maintained and the RCS is not breached. For LOCAs, steamline breaks, and feedline breaks, ECCS injection flow from the RWST maintains long-term subcriticality. This was reflected in the Callaway event tree success 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):

FSAR 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 LCO 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 which is desirable, yet 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 the 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Table 3.1.3-2 of the Callaway IPE notes that boration is not required if hot shutdown conditions are maintained and the RCS is not breached. For LOCAs, steamline breaks, and feedline breaks, ECCS injection flow from the RWST maintains long-term subcriticality. This was reflected in the Callaway event tree success criteria. Further, based on References 6-8, this TS has not been identified as a significant risk contributor. 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):

FSAR 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 include: (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 SDM 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 LCO 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 which desirable, yet 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 3/4.5.4 for Modes 5 and 6; SR 4.5.3.2 addresses Mode 4.

The Callaway IPE does not address initiating events while operating in the shutdown modes.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Table 3.1.3-2 of the Callaway IPE notes that boration is not required if hot shutdown conditions are maintained and the RCS is not breached. For LOCAs, steamline breaks, and feedline breaks, ECCS injection flow from the RWST maintains long-term subcriticality. This was reflected in the Callaway event tree success criteria. Further, based on References 6-8, this TS has not been identified as a significant risk contributor. 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 under 3/4.5.4.

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

FSAR 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 include: (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 LCO 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 which is desirable, yet 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 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Table 3.1.3-2 of the Callaway IPE notes that boration is not required if hot shutdown conditions are maintained and the RCS is not breached. For LOCAs, steamline breaks, and feedline breaks, ECCS injection flow from the RWST maintains long-term subcriticality. This was reflected in the Callaway event tree success criteria. Further, based on References 6-8, this TS has not been identified as a significant risk contributor. 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):

FSAR 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 include, 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 SDM 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 LCO 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 the 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. 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 which is desirable, yet beyond the scope of a primary success path action. 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 Callaway IPE does not address initiating events while operating in the shutdown modes.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Table 3.1.3-2 of the Callaway IPE notes that boration is not required if hot shutdown conditions are maintained and the RCS is not breached. For LOCAs, steamline breaks, and feedline breaks, ECCS injection flow from the RWST maintains long-term subcriticality. This was reflected in the Callaway event tree success criteria. Further, based on References 6-8, this TS has not been identified as a significant risk contributor. 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):

FSAR 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 include, 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 LCO 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 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. 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 which is desirable, yet 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Table 3.1.3-2 of the Callaway IPE notes that boration is not required if hot shutdown conditions are maintained and the RCS is not breached. For LOCAs, steamline breaks, and feedline breaks, ECCS injection flow from the RWST maintains long-term subcriticality. This was reflected in the Callaway event tree success criteria. Further, based on References 6-8, this TS has not been identified as a significant risk contributor. 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):

FSAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.1.3.3 POSITION INDICATING SYSTEMS - SHUTDOWN [APPLICABLE MODES; 3, 4, and 5 with RTBs 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 or 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, this 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 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 equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. 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):

FSAR Chapter 16.

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 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 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.

Parameters with design limits such as SDM, MTC, rod drop time, AFD, F_Q, $F\Delta H$, quadrant power tilt ratio, DNBR, pressurizer and SG pressure and temperature limits are chosen to preclude events from occurring that are non-mechanistically examined in FSAR Chapters 6 and 15. These parameters are not modelled in the PSA which is a best-estimate study of plant design vulnerabilities. As such, these limits are not significant for 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):

FSAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.3.3.2 MOVABLE INCORE DETECTORS
[APPLICABLE MODES; Refer to new FSAR 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Therefore, this TS does not satisfy criterion 4.

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):

FSAR 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Therefore, this TS does not satisfy criterion 4.

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):

FSAR 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. Thus, 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.

The meteorological instrumentation TS has not been shown to be significant to public health and safety by either operating experience or PSA. This instrumentation is not modeled in the Callaway IPE, a Level 2 assessment which does not include an evaluation of offsite dose effects. However, Westinghouse reviewed several Level 3 PSAs in WCAP-11618 and concluded that the requirements of TS 3.3.3.4 are not risk dominant. Offsite dose calculations in these Level 3 PSA studies 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 technical specification to guide emergency measures to protect the public. Routine releases of radioactive materials are not risk significant. Therefore, this technical specification is not of prime importance in risk dominant sequences.

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):

FSAR 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 criterion 3 or 4 are the Type A variables in FSAR Appendix 7A as well as the risk-significant variables listed in the discussion below. Some of the 3/4.3.3.6 instruments may be relocated and some must be retained. Neutron flux and RVLIS will be added.

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 the information needed to perform the required manual actions to bring the plant to a stable condition following an accident. This instrumentation is a component that is part of the primary success path and that 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 meets 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 FSAR 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**

   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**

   Retain for most of the reasons RVLIS is being added.

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 c 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). SG WR level is important in the initiation of feed and bleed cooling which is credited in the Callaway PRA. Several human interactions (HIs) in the PRA are tied to the success of this mode of cooling (IPE Section 3.3.3). Basic event OP-XHE-FO-FANDB, failure to initiate feed and bleed cooling (based on SG WR level in FR-H.1 step 9), has a Risk Achievement Worth (RAW) of 3.07. As discussed in Section 3.3.8 of the IPE, the reported core damage frequency (CDF) is multiplied by the RAW to provide and indication of the effect of a given basic event being unsuccessful. Clearly, the trebling of the CDF is unacceptable. Various SGTR scenarios are also tied to SG WR level. Basic event OP-XHE-FO-SGTRWR, failure to cooldown and depressurize the RCS given a SGTR until after water relief, has a RAW of 5.31. SG WR level would be the best available indication of impending water relief.

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. 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. This instrumentation does not satisfy criterion 3 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. 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. 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). Our exception to installing instrumentation for this variable was approved in NRC's SER dated 4-10-85.

11. **Auxiliary Feedwater Flow Rate**

AFW flow rate should be retained for several reasons:

1. AFW flow rate indication is modeled in the Callaway PRA for cueing operators to restore MFW if AFW is unavailable. Basic event AE-XHE-FO-MFWFLO has a RAW of 1.10 (10% increase in CDF).

2. AFW flow rate indication is vital to the Heat Sink Critical Safety Function (CSF) status tree.

3. Operating experience has proven this indication to be important.

4. AFW flow rate indication is included in NUREG-1431 Table 3.3.3-1.

5. SR 4.7.1.2.1 requires AFW flow rate indication.

6. AFW flow rate indication is being retained in T/S Table 3.3-9 for the ASP. If an LCO and SR for the ASP AFW flow rate indication is being retained, it only makes sense to retain the MCB AFW flow rate indication.

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

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 Callaway 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. It 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. It is a Type D, Category 2 variable and will be relocated.

(4) CONCLUSION

_*_ This Technical Specification is retained.

*As indicated above; neutron flux and RVLIS to be added.

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

FSAR Chapter 16 (containment pressure-extended range, safety valve position indicator, and unit vent-high range noble gas monitor).

PORV 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 3.3.3.8 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 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 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. 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):

FSAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.3.3.10 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 3.3.3.10, 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 3.3.3.10 were deleted in OL Amendment No. 50). 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 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. 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):

FSAR Chapter 16. (The LCO may be relocated but a program statement will be added to the 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 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 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 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. 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):

FSAR 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 3.4.2.1 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 t t 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.

The Callaway IPE does not address initiating events while operating in the shutdown modes. However, based on the risk perspectives gained for the at power sequences reported in the IPE and based on Appendix A, the requirements of this TS are not of prime importance in limiting plant risk. 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):

FSAR 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.

The equipment associated with this TS was modeled in the Callaway Level 2 PSA fault trees; however, based on Appendix A, the requirements of this TS are not of prime importance in limiting plant risk. 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 FSAR 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 specification 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.

The Chemistry Limits governed by this TS, dissolved oxygen, chloride, and fluoride, have little relation to post-accident fission product species of concern (e.g. noble gases, iodine forms, cesium, tellurium, etc.). Refer to Tables 4.7.3-1 and 4.7.3-2 of the Callaway IPE and to the draft NRC source term NUREG-1465. As such, 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):

FSAR 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 3.4.9.2 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.

Parameters with design limits such as SDM, MTC, rod drop time, AFD, F_Q, $F\Delta H$, quadrant power tilt ratio, DNBR, pressurizer and SG pressure and temperature limits are chosen to preclude events from occurring that are non-mechanistically examined in FSAR Chapters 6 and 15. These parameters are not modelled in the PSA which is a best-estimate study of plant design vulnerabilities. As such, these limits are not significant for criterion 4.

(4) CONCLUSION

___ This Technical Specification is retained.

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

FSAR 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.

Loss of component structural integrity is not modeled in the Callaway Level 2 PSA (internal events and flooding only). Our IPEEE program for seismic and fire external events is currently underway. However, failure modes important to risk from an IPEEE would not be identified by this TS. Therefore, this TS does not satisfy criterion 4.

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.

(4) CONCLUSION

___ This Technical Specification is retained.

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

FSAR 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, noncondensible 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.

The equipment associated with this TS was modeled in the Callaway Level 2 PSA fault trees; however, based on Appendix A, the requirements of this TS are not of prime importance in limiting plant risk. 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):

FSAR 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.

Parameters with design limits such as SDM, MTC, rod drop time, AFD, F_Q, $F\Delta H$, quadrant power tilt ratio, DNBR, pressurizer and SG pressure and temperature limits are chosen to preclude events from occurring that are non-mechanistically examined in FSAR Chapters 6 and 15. These parameters are not modelled in the PSA which is a best-estimate study of plant design vulnerabilities. Relocation of this specification does not remove the requirement to perform leak rate testing per 10CFR50 Appendix J. As such, these limits are not significant for 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 FSAR Chapter 16 but the limiting values of P_a and L_a will be retained in the CONTAINMENT INTEGRITY Bases. Relocation of LCO 3.6.1.2 requires that revisions be made to SR 4.6.1.1.c 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 specification 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA. PRAs indicate that risk is dominated by events in which the containment is bypassed, unisolated, or fails structurally. Containment failure frequency is determined by comparing containment failure pressure. As discussed in the Callaway IPE Section 4.4.2, the containment failure pressure is based on realistic material properties as well as conservative calculations of containment stress. The material properties used in these calculations do not change rapidly, so the 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):

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

TS6F

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 an 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.

Parameters with design limits such as SDM, MTC, rod drop time, AFD, F_Q, $F\Delta H$, quadrant power tilt ratio, DNBR, pressurizer and SG pressure and temperature limits are chosen to preclude events from occurring that are non-mechanistically examined in FSAR Chapters 6 and 15. These parameters are not modelled in the PSA which is a best-estimate study of plant design vulnerabilities. As such, these limits are not significant for criterion 4.

(4) CONCLUSION

___ This Technical Specification is retained.

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

FSAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.7.8 SNUBBERS
[APPLICABLE MODES; 1, 2, 3, and 4 - also 5 and 6 for snubbers on systems required to be operable in 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 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 Snubber Surveillance Program at Callaway.

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, the snubber requirements do not satisfy criterion 3.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. 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):

FSAR 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.

Parameters with design limits such as SDM, MTC, rod drop time, AFD, F_Q, $F\Delta H$, quadrant power tilt ratio, DNBR, pressurizer and SG pressure and temperature limits are chosen to preclude events from occurring that are non-mechanistically examined in FSAR Chapters 6 and 15. These parameters are not modelled in the PSA which is a best-estimate study of plant design vulnerabilities. As such, these limits are not significant for criterion 4.

(4) CONCLUSION

___ This Technical Specification is retained.

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

FSAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.7.12 AREA TEMPERATURE MONITORING [APPLICABLE MODES; Whenever equipment in 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 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, the TS is only indirectly applicable to the operability of these systems and components. Therefore, this TS does not satisfy criterion 3.

Although room heatup calculations were reviewed during the Callaway IPE to determine equipment survivability, the normal operation limits governed by this TS have only a secondary relationship to post-accident and off-normal room temperatures and no relation to the EQ test data used to determine equipment functionality. In general, room coolers were determined to be risk significant; however, initial room conditions are not overly important. As such, 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):

FSAR 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 3.8.4.1 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 equipment associated with this TS was not modeled in the Callaway Level 2 PSA. Failure of containment building penetrations (electrical, fluid, equipment hatch, personnel hatch, etc.) was evaluated in the NUREG-1150 analysis and was judged to be significantly less important than over-pressure failure of the cylinder wall. Callaway-specific analyses examined the possible failure of the containment in the locale of large pipe penetrations and surrounding liner in the containment mid-height region, large pipe penetrations at the wall/base slab interface, small pipe and electrical penetrations, and the containment access systems (equipment hatch, personnel lock, and escape lock). The large pipe penetrations at the mid-height region and the containment access systems were identified as weak links; however, the predicted lower bound capacity of these areas is significantly higher than the critical weak link, failure in the mid-height region of the containment wall.

The electrical penetrations at Callaway are supplied with nitrogen which pressurizes the cavity between the seals. While the electrical penetrations are considered to be safety-related, the nitrogen supply is not. The potential for degradation of the seal mechanism due to overpressurization or due to loss of the nitrogen supply has been evaluated. Multiple failures would be required to supply nitrogen to the penetration assemblies at over 300 psig. Plant evaluations have identified that pressures up to 400 psig would not result in failures of the seals. Therefore, overpressurization of the seals is not a credible scenario. The design of the electrical penetration assemblies is such that the nitrogen charge is provided to facilitate early detection of seal degradation and to provide an inert atmosphere to inhibit internal corrosion of the assemblies. The nitrogen charge is not required for the electrical penetrations to maintain their containment isolation function. Based on these considerations, the nitrogen system is not essential to maintaining containment integrity in the long term. 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):

FSAR Chapter 16.

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 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 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. 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):

FSAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.9.6 REFUELING MACHINE
[APPLICABLE MODES; During movement of drive rods or fuel assemblies within RV.]

(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 equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. 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):

FSAR Chapter 16.

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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. 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):

FSAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATT'.N 3.9.10.2 WATER LEVEL - REACTOR VESSEL/CONTROL RODS
[APPLICABLE MODES; 6 during movement of control rods within the RV.]

(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 water level - 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 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.

Parameters with design limits such as SDM, MTC, rod drop time, AFD, F_Q, $F\Delta H$, quadrant power tilt ratio, DNBR, pressurizer and SG pressure and temperature limits are chosen to preclude events from occurring that are non-mechanistically examined in FSAR Chapters 6 and 15. These parameters are not modelled in the PSA which is a best-estimate study of plant design vulnerabilities. As such, these limits are not significant for criterion 4.

(4) CONCLUSION

___ This Technical Specification is retained.

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

FSAR 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 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 Specification 3.10.1 will be deleted.

Parameters with design limits such as SDM, MTC, rod drop time, AFD, F_Q, $F\Delta H$, quadrant power tilt ratio, DNBR, pressurizer and SG pressure and temperature limits are chosen to preclude events from occurring that are non-mechanistically examined in FSAR Chapters 6 and 15. These parameters are not modelled in the PSA which is a best-estimate study of plant design vulnerabilities. As such, these limits are not significant for 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.

Parameters with design limits such as SDM, MTC, rod drop time, AFD, F_Q, $F\Delta H$, quadrant power tilt ratio, DNBR, pressurizer and SG pressure and temperature limits are chosen to preclude events from occurring that are non-mechanistically examined in FSAR Chapters 6 and 15. These parameters are not modelled in the PSA which is a best-estimate study of plant design vulnerabilities. As such, these limits are not significant for criterion 4.

(4) CONCLUSION

___ This Technical Specification is retained.

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

FSAR 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 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Accidents evaluated in FSAR Section 15.7 (other than FHA) result in insignificant offsite dose consequences when compared either to the design basis LBLOCA or to the beyond design basis scenarios examined in the Callaway IPE (e.g. scenarios with corium released from a breached reactor vessel, etc.). 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):

FSAR 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 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 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 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 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 does not satisfy criterion 3.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Accidents evaluated in FSAR Section 15.7 (other than FHA) result in insignificant offsite dose consequences when compared either to the design basis LBLOCA or to the beyond design basis scenarios examined in the Callaway IPE (e.g. scenarios with corium released from a breached reactor vessel, etc.). 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):

FSAR 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.

The equipment associated with this TS was not modeled in the Callaway Level 2 PSA nor is it known to be significant based on risk insights from other PSAs or operating experience. Accidents evaluated in FSAR Section 15.7 (other than FHA) result in insignificant offsite dose consequences when compared either to the design basis LBLOCA or to the beyond design basis scenarios examined in the Callaway IPE (e.g. scenarios with corium released from a breached reactor vessel, etc.). 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):

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

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 Callaway Technical Specifications have been evaluated based on those four criteria. If none of the first three criteria was identified as a constraint on the Technical Specification in question, then the Technical Specification was identified as a candidate for relocation to another controlled document. The purpose of this task is to determine if the parameters, components, or systems addressed by the relocation candidate 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 that are relocation candidates is based on the following:

A. The Technical Specifications that are relocated will be transferred to FSAR Chapter 16 and will be implemented by programs and procedures subject to control by Union Electric, 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 FSAR Chapter 16.

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 FSAR Chapter 16.

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 Callaway IPE (Reference 1) is considered to be a dominant sequence. This is roughly 2% of the total Callaway core damage frequency, 5.846E-05 per reactor year. In addition, any sequence whose frequency of containment bypass was found to be greater than 1.0E-07 per reactor year in the Callaway IPE is also considered to be a dominant sequence. These sequences are identified in Section 3.4 of the Callaway IPE.

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 Callaway IPE is considered to be a dominant sequence. This criterion is consistent with the present NRC position in the Safety Goal Policy, i.e., a goal of 1E-06 per reactor year for the total frequency of a severe offsite release and no greater than 1E-07 per reactor year for an individual sequence. As discussed in Section 4.7 of the Callaway IPE, accident sequences with similar release characteristics were grouped into source term categories (STCs). As further shown in Tables 4.7.2-1 and 4.7.3-2 of the Callaway IPE, only STCs 1, 9, and 10 are associated with severe releases that have composite frequencies (for all their associated sequences) greater than 1.0E-07 per reactor year. These STCs are all early releases (<24 hours) due to containment bypass. STC 1 is associated with an interfacing systems LOCA in the auxiliary building. STCs 9 and 10 are associated with SGTRs; STC 9 has a release path via a stuck open SG atmospheric relief valve while STC 10's release path is through the condenser. The accident sequences associated with these STCs are the same three that meet the containment bypass criterion above and are discussed in Sections 4.13-4.15 below.

The risk dominant sequences meeting the above criteria are discussed in Section 4.0.

E. Callaway systems and functions that are important from a PSA or operating experience perspective are listed in Table 1. These identified systems, as well as the risk dominant sequences from the Callaway IPE, were used to screen the requirements of the Technical Specifications reviewed. Those Technical Specifications whose requirements were relevant to these systems and sequences were further evaluated for risk dominance. If the requirements of a Technical Specification were not found to be modeled in the Callaway IPE and no significant issues were identified from a review of the risk insights or operating experience, that Technical Specification would be relocated to FSAR Chapter 16 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:

- The number and title of the Technical Specification;
- An evaluation of the Technical Specification against the Final Policy Statement's four criteria;
- A discussion of the information that was used as the basis to arrive at the conclusions for the four criteria; and
- 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 2).

4.0 RISK DOMINANT SEQUENCES

In the discussions that follow, sequence cutsets are not provided for flooding or interfacing systems LOCA sequences since the methodologies used to calculate the CDF in these areas of the Callaway IPE do not generate cutsets (see Section 3.4.1.3 of Reference 1).

4.1 Internal Flooding Event: Flood Zone 2, Sequence No. 5 (FLOODS05)

Loss of ESW and SW, Leading to Core Damage

Sequence Descriptor: FLOODSWESW

Core Damage Frequency (CDF): 9.9E-06/yr

Description:

This scenario involves flooding in Room 3101 in the basement of the control building (elevation 1974 ft.). The valves which isolate the Essential Service Water (ESW) system from the Service Water (SW) system are located in this room. Other piping systems, such as fire water (FW) and domestic water, pass through this room to other areas of the control building. Important PSA equipment include the cross-tie valves between the ESW and SW systems. Postulated pipe breaks in these systems were evaluated over a spectrum of break sizes and provide the initiating internal flooding source. Flooding to the height of the valve motor operators would make it impossible for the control room staff to correctly align the ESW system for operation and isolate the Service Water system. The most critical factor then becomes the ability to gain access to the room and manually close the valves, as well as provide additional drainage to the area. The design basis calculations for flooding in this area assume a fire water break, allowing an equipment operator 21 minutes for response. Depending on the size of the break for all systems considered, the internal flooding study calculates from 3 to 15 minutes for operator response. The consequence of a non-recoverable total loss of all service water (both ESW and SW) would ultimately result in a reactor coolant pump loss of seal cooling and failure of the high and intermediate head ECCS pumps. This would lead to core damage.

4.2 Station Blackout Event Tree, Sequence No. 03 (T(1S)S03)

Sequence Descriptor: T(1S)D(1)

CDF: 6.969E-06/yr

Description:

This core damage sequence is a station blackout, followed by successful decay heat removal via the turbine driven auxiliary feedwater pump, successful cooldown and depressurization of the reactor coolant system (in order to minimize leakage from the RCP seals due to loss of all seal cooling), AC (offsite) power recovery within 8 hours (prior to core uncovery), but subsequent failure of the high head (CCPs) and intermediate head (SI) ECCS pumps to inject (in order to make up inventory to mitigate the RCP seal LOCA). This would result in core uncovery and, therefore, core damage.

Sequence Cutsets:

In general, the sequence cutsets reflect a loss of offsite power followed by failure of the diesel generators either directly (fail to run, fail to start, test/maintenance) or indirectly (ESW failures, for example), power recovery within 8 hours, core not uncovered, and failure of the centrifugal charging pumps (CCPs) and safety injection pumps (SIPs). Failure of the CCPs/SIPs is caused by inability of the operators to actuate the pumps (represented by basic event "OP-XHE-FO-ACRECV") or by support system failures. If, in the latter case, the CCPs/SIPs were failed due to failure of ESW, credit is taken for the availability of the SW system to cool ESW heat loads after AC (offsite) power recovery. This "credit" was applied to selected cutsets as a recovery factor, and is represented by the basic event "EA-XHE-FO-SWSBO". This basic event includes both the human error probability and the hardware failure probability associated with recovering the SW system in these scenarios.

4.3 Internal Flooding Event: Flood Zone 3, Sequence No. 4 (FLOODS04)

Loss of Vital AC Buses, Leading to Core Damage

Sequence Descriptor: FLOODACBUS

CDF: 5.600E-06/yr

Description:

This scenario involves flooding of the ESF Switchgear Rooms 3301 and 3302 in the control building (elevation 2000 ft.). These rooms are adjacent to the diesel generator rooms and contain critical electrical equipment. Important PSA equipment include the vital 4.16 kVac buses NB01 and NB02, 480 Vac load centers NG01, NG02, NG03, and NG04, and 480 Vac MCCs NG01A and NG02A. Essential service water and firewater lines are routed in these areas and a spectrum of postulated pipe breaks in these systems provides the initiating internal flooding source. Flooding is only of concern when both switchgear rooms are flooded and the two trains of vital AC are disabled. Flooding to critical heights in these cases include flooding of both switchgear rooms, either by failure of connecting doors or under doorways and through gaps. Flooding of the adjacent diesel generator rooms and propagation of this flood to the switchgear rooms is also postulated. For a flood originating in a switchgear room, the times for operator response to terminate the flood and provide drainage to the area range from 2 hours to 6 minutes.

For a flood originating in a diesel generator room that propagates to the switchgear rooms, the times for operator response range from 1.5 hours to 14 minutes. The consequence of a non-recoverable total loss of all AC power (station blackout) would result in a reactor coolant pump seal loss of cooling and failure of the high and intermediate head ECCS pumps, leading to core damage.

4.4 Station Blackout Event Tree, Sequence No. 26 (T(1S)S26)

Sequence Descriptor: T(1S)L(2)E(1)

CDF: 4.738E-06/yr

Description:

This core damage sequence is a station blackout, followed by failure of the turbine driven auxiliary feedwater pump (TDAFP) and failure to recover AC power within 1 hour. This results in core uncovery due to loss of the secondary heat sink.

Note that the time to core uncovery in this scenario would actually be approximately 2 hours (1 hour for steam generator dryout and 1 hour for uncovery to the top of the core). However, for convenience, 1 hour was used in the event tree. This was done so that the pressurizer PORV success criterion used in the upper event tree branch (success of the TDAFP) could be used in this branch (failure of the TDAFP). Also, the probabilities of failure to recover AC power in 1 hour and 2 hours are the same order of magnitude.

Sequence Cutsets:

In general, the sequence cutsets are comprised of a loss of offsite power, followed by failure of the diesel generators either directly or indirectly, direct failure of the TDAFP (test/maintenance, fail to start), and failure to recover AC power within 1 hour.

4.5 Small LOCA Event Tree, Sequence No. 03 (S(2)S03)

Sequence Descriptor: S(2)H(1)H(3)

CDF: 3.885E-06/yr

Description:

This core damage sequence is a small LOCA, followed by a reactor trip, successful inventory makeup via high and intermediate head ECCS pump operation in the injection mode, successful decay heat removal using the auxiliary feedwater pumps, but failure of the high and intermediate head ECCS pumps in the cold leg recirculation mode; and following successful RCS cooldown and depressurization, failure of RHR system operation in the recirculation mode. The failure of all ECCS pumps results in an inability to supply inventory makeup to the RCS, which results in core uncovery and subsequent core damage.

Sequence Cutsets:

The sequence cutsets for a small LOCA reflect the same types of failures as the sequence cutsets for an intermediate LOCA. See the cutset description for sequence S(1)S03 below.

4.6 Intermediate LOCA Event Tree, Sequence No. 03 (S(1)S03)

Sequence Descriptor: S(1)H(1)H(3)

CDF: 3.876E-06/yr

Description:

This core damage sequence is an intermediate LOCA, followed by successful inventory makeup via operation of the high and intermediate head ECCS pumps in the injection mode. However, subsequent failure of the high and intermediate head ECCS pumps in the cold leg recirculation mode requires the cooldown and depressurization of the RCS to allow the operation of the RHR system. This is accomplished via successful operation of the auxiliary feedwater pumps to remove decay heat and use of either the primary PORVs, the steam generator (SG) atmospheric relief valves, or the steam dumps to depressurize. However, after cooldown and depressurization of the RCS is completed, failure of both trains of the RHR system to operate in the cold leg recirculation mode occurs. The failure of all ECCS pumps in cold leg recirculation results in an inability to supply makeup to the RCS, which results in core uncovery and subsequent core damage.

Sequence Cutsets

In general, the sequence cutsets reflect an intermediate LOCA followed by failure of the RHR system which results in an inability to provide flow to the RCS either directly from the RHR pumps or via the high and intermediate head ECCS pumps which depend upon the RHR pumps for suction flow when in the cold leg recirculation mode. The failure of the RHR system is either caused directly or indirectly. The direct failures are characterized by component failures such as pump failures to start or run, by system test or maintenance outages, or by human errors in aligning CCW flow to the RHR heat exchangers or in separating the RHR trains during cold leg recirculation (the RHR trains are cross-connected during injection). The indirect failures that result in RHR system failure are characterized by various faults in the ESW, CCW, and RHR pump room cooling systems, as well as in the actuation signal for the switchover to cold leg recirculation. Included in the acutation signal faults is a human error for the miscalibration of the RWST level instrumentation which provides the automatic switchover signal to the RWST and recirculation sump suction isolation valves.

4.7 Transient-Induced Loss of RCP Seal Cooling Event Tree, Sequence No. 02 (TRCPS02)

Sequence Descriptor: TRCPH(3)

CDF: 3.096E-06/yr

Description:

This core damage sequence is a summation of transient-induced losses of RCP seal cooling caused by loss of offsite power, loss of a DC bus, loss of main feedwater, and turbine trip with main feedwater available. Because the core damage frequency is dominated by the cutsets resulting from a loss of offsite power (2.871E-06/yr), which are an order of magnitude higher than the next highest sequence cutsets, the loss of offsite power case will be discussed.

This core damage sequence is a loss of offsite power followed by a reactor trip, successful restoration of power to at least one NB bus via operation of its associated diesel generator, successful closure of any open pressurizer PORV, but with a subsequent failure to maintain RCP seal cooling which leads to a seal LOCA and the loss of seal cooling event tree. The loss of seal cooling is followed by successful removal of decay heat via the auxiliary feedwater pumps, successful cooldown and depressurization of the RCS, successful RCS inventory makeup via the safety injection accumulators and RHR injection, but with subsequent failure of RHR cold leg recirculation. The failure of RHR cold leg recirculation results in an inability to provide makeup for the inventory lost via the RCP seal LOCA, which results in core uncovery and core damage.

Sequence Cutsets:

In general, the sequence cutsets reflect a loss of offsite power followed by a failure of a diesel generator and a failure of the opposite train ESW system which results in a loss of both trains of the RHR system. Credit was taken for operator action to restore failures of ESW valves prior to either loss of the remaining diesel generator or failure of the RHR train and was applied to selected cutsets to mitigate the effects of these failures. In addition, credit was taken for AC power recovery in 8 hours to mitigate ESW failures that affect the CCW system. This was based on an evaluation that there would be approximately 8 hours before the switchover to RHR recirculation would be required. This recovery action was applied to selected cutsets to mitigate the effects of these failures.

4.8 Loss of All Service Water Event Tree, Sequence No. 04 (T(SW)S04)

Sequence Descriptor: T(SW)SW2SW8

CDF: 2.863E-06/yr

Description:

This core damage sequence is a complete loss of service water (including ESW), followed by a reactor trip, successful decay heat removal via the turbine driven auxiliary feedwater pump, an inability to recover service water within 2 hours, successful RCS cooldown and depressurization, successful inventory makeup using the safety injection accumulators and the RHR system operating in the injection mode, but the inability to recover the service water system in 8 hours. The inability to recover service water at either 2 or 8 hours results in an RCP seal LOCA due to an inability to operate either the centrifugal charging pumps or the CCW system, followed by an inability to operate the RHR system in the cold leg recirculation mode following depletion of the RWST. The inability to mitigate the RCP seal LOCA results in core uncovery and subsequent core damage.

Sequence Cutsets:

The sequence cutsets are comprised of a complete loss of service water and an inability to recover either service water or ESW flow in 2 or 8 hours due to either human errors or equipment failures that are not recoverable in 2 or 8 hours.

4.9 Station Blackout Event Tree, Sequence No. 10 (T(1S)S10)

Sequence Descriptor: T(1S)E(8)E(X)

CDF: 2.571E-06/yr

Description:

This core damage sequence is a station blackout, followed by successful operation of the turbine driven auxiliary feedwater pump (TDAFP) and successful RCS cooldown and depressurization; however, AC power is not recovered in 12 hours. The station batteries will fail at 8 hours, resulting in TDAFP failure. With RCS cooldown and depressurization, core uncovery will occur at approximately 12 hours in this scenario, if AC power is not recovered, due to the loss of secondary heat sink.

Sequence Cutsets:

In general, the sequence cutsets are comprised of a loss of offsite power, failure of the diesel generators either directly (fail to run, fail to start, test/maintenance) or indirectly (i.e., ESW failures), failure to recover AC power within 8 hours, and (conditional) failure to recover AC power in 12 hours.

4.10 Internal Flooding Event: Flood Zone 4, Sequence No. 3 (FLOODS03)

Loss of all DC Power, Leading to Core Damage

Sequence Descriptor: FLOODDCBUS

CDF: 2.300E-06/yr

Description:

This scenario involves flooding of the rooms and corridors in the control building on the 2016 ft. elevation. This area includes primarily DC battery and inverter rooms and connecting corridors. Important PSA equipment include 125 Vdc buses NK01, NK02, NK03, and NK04. Essential service water and firewater lines are primarily routed in the corridors and a flood is postulated to propagate to the battery rooms. A spectrum of postulated pipe breaks provides the initiating internal flooding source. Floods originate in the connecting corridors and propagate to the battery rooms by leaking through gaps and under doors. Without intervention, this flood would reach the critical height within approximately 8 minutes. No credit is given for operator action since the response time required is short and there would be no clear indication in the control room as to the nature of the problem, only where water is collecting. The consequence of a non-recoverable total loss of all DC power has not been analyzed. Even though accident mitigation measures may be possible without DC power, the ability to respond to degraded plant conditions without instrumentation is highly uncertain. For conservatism this scenario has been assumed to lead to core damage.

4.11 Large LOCA Event Tree, Sequence No. 02 (AS02)

Sequence Descriptor: AH(3)

CDF: 1.819E-06/yr

Description:

This core damage sequence is a large LOCA, followed by successful inventory makeup via the safety injection accumulators and RHR system injection but failure of the RHR system to operate in the cold leg recirculation mode. This results in core uncovery due to insufficient makeup, and thus core damage.

Sequence Cutsets:

The sequence cutsets reflect a large LOCA and the failure of both trains of the RHR system in the cold leg recirculation mode, either directly or indirectly. The direct causes of RHR system failure are human errors in aligning CCW to the RHR heat exchangers or in separating the two trains of RHR (the RHR trains are cross-connected during injection), pump failures to start or run, and test and maintenance outages. The indirect causes of RHR system failure include failures of the support systems, i.e. ESW, CCW, RHR pump room cooling, and the RWST level signals that provide the input for the automatic switchover to cold leg recirculation. The switchover actuation signal failure also includes a human error for the miscalibration of the RWST level instruments which provide the input to the signal for automatic switchover from ECCS injection to cold leg recirculation.

4.12 Station Blackout Event Tree, Sequence No. 04 (T(1S)S04)

Sequence Descriptor: T(1S)C(UN)

CDF: 1.513E-06/yr

Description:

This core damage sequence is a station blackout, followed by successful operation of the turbine driven auxiliary feedwater pump, successful RCS cooldown and depressurization, AC power recovery within 8 hours, but the core is uncovered (due to the RCP seal LOCA with no makeup) when AC power is restored.

Sequence Cutsets:

In general, the sequence cutsets are comprised of a loss of offsite power, followed by failure of the diesel generators either directly (fail to start, fail to run, test/maintenance) or indirectly (i.e., ESW failure), AC power recovery within 8 hours, but the core is uncovered when power is recovered.

4.13 Steam Generator Tube Rupture Event Tree, Sequence No. 07 (T(SG)S07)

Sequence Descriptor: T(SG)O(2)O(3)

CDF: 3.656E-07/yr

Description:

This core damage sequence is a steam generator tube rupture followed by a reactor trip, successful inventory makeup via the operation of the high and intermediate head ECCS pumps in the injection mode, and successful isolation of the ruptured steam generator. However, subsequent to these successful operations, there are failures to cooldown and depressurize the RCS either before or after water relief from the SG atmospheric relief valves and/or safety valves. This results in a release of radioactive material bypassing the containment, and if the release is allowed to continue for an extended period, the eventual uncovery of the core with resultant core damage.

Sequence Cutsets:

The sequence cutsets reflect a steam generator tube rupture and an inability to cooldown and depressurize the RCS either before or after water relief due to either human error or failures of the primary PORVs or the SG atmospheric relief valves and the steam dumps.

4.14 Steam Generator Tube Rupture Event Tree, Sequence No. 09 (T(SG)S09)

Sequence Descriptor: T(SG)Q(PS)E(CA)

CDF: 3.387E-07/yr

Description:

This core damage sequence is a steam generator tube rupture followed by a reactor trip, successful RCS inventory makeup via high and intermediate head ECCS pump injection, successful decay heat removal via the auxiliary feedwater pumps, but with a subsequent failure to isolate the ruptured steam generator and a failure to perform a long term cooldown and depressurization of the RCS. This results in a release of radioactive material that bypasses the containment and the eventual uncovery of the core with resultant core damage due to an inability to provide RCS inventory makeup.

Sequence Cutsets:

In general, the sequence cutsets are comprised of a steam generator tube rupture coupled with failures of the affected SG's main steam isolation valve due to either component failure or human error, and failures of the RHR system. The RHR system failures are either direct failures due to human error or component failure, or indirect failures due to failures in support systems such as ESW, CCW, or AC power.

4.15 Interfacing Systems LOCA, RHR Suction Line Rupture, Sequence No. 6 (ILOCAS06)

Sequence Descriptor: ILOCARHRSC

CDF: 1.010E-07/yr

Description:

This core damage sequence begins with overpressurization of the RHR suction line. The principal cause of overpressurization is a common cause failure of the pressure isolation valves in one of the two RHR suction lines. The RHR suction lines tap off the RCS hot legs in loops 1 and 4. Each suction line has two pressure isolation MOVs in series. Over 77% of the initiating event frequency is attributed to this failure mechanism. Once overpressurized, the RHR system may rupture. While piping, valves, pumps, gaskets, and heat exchangers were reviewed, it was determined that the RHR heat exchangers were the most likely equipment to fail (rupture frequency = 0.427%). If this rupture were to occur, core damage would result if the operator failed to cooldown and depressurize the RCS and refill the RWST. This sequence is a containment bypass sequence.

5.0 REFERENCES

1. ULNRC-2703 dated September 29, 1992, Callaway Plant Individual Plant Examination.

2. WCAP-11618, "Methodically Engineered, Restructured and Improved Technical Specifications," MERITS Program-Phase II, Task 5, Criteria Application.

3. Letter from T. E. Murley (NRC-NRR) to W.S. Wilgus (B&W Owners Group Chairman) dated May 9, 1988, Appendix B, Results of the NRC Staff Review of the WOG's Submittal on Retention and Relocation of Specific Technical Specifications.

4. 58FR39132 dated July 22, 1993, NRC's Final Policy Statement on Technical Specification Improvements.

TABLE 1

FRONT-END ANALYSIS[1]	BACK-END ANALYSIS[1]
Essential service water	Containment heat removal (RHR and containment fan coolers)
Service water[2]	Containment spray
Component cooling water	SGTR recovery[7]
ECCS[3]	
RCP seal cooling (CVCS, CCW)	
Auxiliary feedwater (especially the turbine-driven AFW pump)	
Class 1E AC/DC systems[4]	
Offsite power[2]	
Feed and bleed cooling (high pressure ECCS injection and pressurizer PORVs)	
SG atmospheric relief valves (ARVs)	
RWST	
RHR interfacing systems isolation valves	
Nitrogen accumulators for AFW and SG ARVs	
Main feedwater (including condensate pumps)[2]	
Room coolers[5]	
Reactor trip system[6]	
Solid state protection system[6]	
7300 process protection system[6]	
BOP-ESFAS[6]	

Notes:

1) As defined in NUREG-1335 and used in the Callaway IPE (a Level II PSA).
2) Non-safety system found to have risk significance.
3) Includes SI accumulators, centrifugal charging pumps, safety injection pumps, and RHR pumps.
4) Includes 4.16 kVac buses and diesel generators, 480 Vac load centers and MCCs, 120 Vac switchboards, 125 Vdc batteries and panels, LOCA and shutdown sequencers.
5) Includes AFW, CCP, SI, RHR, CCW, and containment spray pump room coolers and the DG ventilation supply fans.
6) Includes all circuitry needed to generate reactor trip, SIS, CIS-A, CIS-B, containment spray, auxiliary feedwater, feedwater isolation, steamline isolation, containment purge isolation, RWST low-low-1 switchover, as well as the actuated end devices.
7) Includes MSIVs, SG atmospheric relief valves (steam dumps if condenser is available), pressurizer PORVs, and RHR.

ATTACHMENT SIX

DRAFT FSAR CHAPTER 16 MARK-UPS

CHAPTER 16.0

TECHNICAL SPECIFICATIONS

See the Callaway Plant Technical Specifications (NUREG-1058), Appendix A to NRC License No. NPF-30, for the retained specifications. Specifications contained in this chapter have been relocated in accordance with the NRC Final Policy Statement on Technical Specification Improvements, 58FR39132 dated July 22, 1993.

16.1 (3/4.1) REACTIVITY CONTROL SYSTEMS

16.1.1 INTENTIONALLY BLANK

16.1.2 (3/4.1.2) BORATION SYSTEMS

16.1.2.1 (3.1.2.1) LIMITING CONDITION FOR OPERATION

FLOW PATH - SHUTDOWN

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.

16.1.2.1.1 (4.1.2.1) SURVEILLANCE REQUIREMENTS

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.

16.1.2.1.2 BASES

The Boration Systems ensure 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 functional capability in the event an assumed single 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 either 17,658 gallons of 7000 ppm borated water from the boric acid storage tanks or 83,745 gallons of 2350 ppm borated water from the RWST. With the RCS average temperature less than 350°F, 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 boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ 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 14,076 gallons of 2350 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.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.4 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.

16.1.2.2 LIMITING CONDITION FOR OPERATION
(3.1.2.2)

FLOW PATHS - OPERATING

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. (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 prior to the temperature of one or more of the RCS cold legs exceeding 375°F.)

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% $\Delta k/k$ 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.

16.1.2.2.1 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 Section 16.1.2.2a delivers at least 30 gpm to the Reactor Coolant System.

16.1.2.2.2 BASES

See Section 16.1.2.1.2.

16.1.2.3 LIMITING CONDITION FOR OPERATION
(3.1.2.3)

CHARGING PUMP - SHUTDOWN

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.

16.1.2.3.1 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 Technical Specification 4.0.5.

16.1.2.3.2 BASES

See Section 16.1.2.1.2.

16.1.2.4 LIMITING CONDITION FOR OPERATION
(3.1.2.4)

CHARGING PUMPS - OPERATION

At least two centrifugal charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3. (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 prior to the temperature of one or more of the RCS cold legs exceeding 375°F.)

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 centrifugal charging pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

16.1.2.4.1 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 Technical Specification 4.0.5.

16.1.2.4.2 BASES

See Section 16.1.2.1.2.

16.1.2.5 LIMITING CONDITION FOR OPERATION
(3.1.2.5)

BORATED WATER SOURCE - SHUTDOWN

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 2350 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.

16.1.2.5.1 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.

16.1.2.5.2 BASES

See Section 16.1.2.1.2.

16.1.2.6 LIMITING CONDITION FOR OPERATION
(3.1.2.6)

BORATED WATER SOURCES - OPERATING

As a minimum, the following borated water sources shall be OPERABLE as required by 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 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 2350 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% $\Delta k/k$ 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.

16.1.2.6.1 SURVEILLANCE REQUIREMENTS
(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.

16.1.2.6.2 <u>BASES</u>

See Section 16.1.2.1.2.

16.1.3 <u>MOVABLE CONTROL ASSEMBLIES</u>
(3/4.1.3)

16.1.3.1 <u>LIMITING CONDITION FOR OPERATION</u>
(3.1.3.3)

<u>POSITION INDICATION SYSTEM-SHUTDOWN</u>

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.

<u>APPLICABILITY</u>: MODES 3, 4, and 5. (With the Reactor Trip System breakers in the closed position. See Special Test Exception in Section 16.10.1.)

<u>ACTION</u>:

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

16.1.3.1.1 <u>SURVEILLANCE REQUIREMENTS</u>
(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.

16.1.3.1.2 <u>BASES</u>

See Technical Specification Bases 3/4.1.3.

16.1.3.2 LIMITING CONDITION FOR OPERATION
(3.1.3.4)

ROD DROP TIME

The individual full-length shutdown and control rod drop time from the 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.

16.1.3.2.1 SURVEILLANCE REQUIREMENTS
(4.1.3.4)

See Technical Specification 4.1.3.1.3.

16.1.3.2.2 BASES

See Technical Specification Bases 3/4.1.3.

16.3.1.5 LIMITING CONDITION FOR OPERATION
(3.3.3.8)

LOOSE-PART DETECTION SYSTEM

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.

16.3.1.5.1 SURVEILLANCE REQUIREMENTS
(4.3.3.8)

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.

16.3.1.5.2 BASES

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.

16.3.1.6 LIMITING CONDITION FOR OPERATION
(3.3.3.10)

WASTE GAS HOLDUP SYSTEM

EXPLOSIVE GAS MONITORING INSTRUMENTATION

At least one hydrogen and both the inlet and outlet oxygen explosive gas monitoring instrument channels for each WASTE GAS HOLDUP SYSTEM recombiner shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Section 16.11.2.1 are not exceeded.

16.2 INTENTIONALLY BLANK

16.3 (3/4.3) INSTRUMENTATION

16.3.1 (3/4.3.3) MONITORING INSTRUMENTATION

16.3.1.1 (3.3.3.2) LIMITING CONDITION FOR OPERATION

MOVABLE INCORE DETECTORS

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, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$, and F_{xy}.

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.

16.3.1.1.1 (4.3.3.2) SURVEILLANCE REQUIREMENTS

The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours be 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 $F^N_{\Delta H}$, $F_Q(Z)$, and F_{xy}.

16.3.1.1.2 BASES

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 $F_Q(Z)$ or $F^N_{\Delta H}$ 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.

16.3.1.2 LIMITING CONDITION FOR OPERATION
(3.3.3.3)

SEISMIC INSTRUMENTATION

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.

16.3.1.2.1 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.

16.3.1.2.2 BASES

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.

16.3.1.3 LIMITING CONDITION FOR OPERATION
(3.3.3.4)

METEOROLOGICAL INSTRUMENTATION

The meteorological monitoring instrumentation channels shown 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 6.9.2 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.

16.3.1.3.1 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 16.3-4.

16.3.1.3.2 BASES

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are 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.

16.3.1.4 LIMITING CONDITION FOR OPERATION
(3.3.3.6)

ACCIDENT MONITORING INSTRUMENTATION

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 requirement of Table 16.3-5, 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 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 channel to OPERABLE status.

d. The provisions of Technical Specification 3.0.4 are not applicable.

16.3.1.4.1 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 16.3-6.

16.3.1.4.2 BASES

See Technical Specification Bases 3/4.3.3.6.

APPLICABILITY: During WASTE GAS HOLDUP SYSTEM operation.

ACTION:

a. With an outlet oxygen monitor channel inoperable, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours.

b. With both oxygen or both hydrogen channels or both the inlet oxygen and inlet hydrogen monitor channels for one recombiner 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.

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

16.3.1.6.1 (4.3.3.10) SURVEILLANCE REQUIREMENTS

Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of:

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

b. An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 92 days with the use of standard gas samples containing a nominal:

1) One volume percent hydrogen, balance nitrogen and four volume percent hydrogen, balance nitrogen for the hydrogen monitor, and

2) One volume percent oxygen, balance nitrogen, and four volume percent oxygen, balance nitrogen for the inlet oxygen monitor, and

3) 10 ppm by volume oxygen, balance nitrogen and 80 ppm by volume oxygen, balance nitrogen for the outlet oxygen monitor.

16.3.1.6.2 INTENTIONALLY BLANK

16.3.2 (3/4.3.4) TURBINE OVERSPEED PROTECTION

16.3.2.1 (3.3.4) LIMITING CONDITION FOR OPERATION

At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3. (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.)

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.

16.3.2.1.1 SURVEILLANCE REQUIREMENTS

16.3.2.1.1.a
(4.3.4.1)

The provisions of Technical Specification 4.0.4 are not applicable.

16.3.2.1.1.b
(4.3.4.2)

The above required Turbine Overspeed Protection System shall be maintained, calibrated, tested, and inspected in accordance with the Callaway Plant's Turbine Overspeed Protection Reliability Program. Adherence to this program shall demonstrate OPERABILITY of this system. The program and any revisions should be reviewed and approved in accordance with Technical Specification 6.5.1.6o. Revisions shall be made in accordance with the provisions of 10 CFR 50.59.

16.3.2.1.2 BASES

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.

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	± 1.0 g	1
b. Control Room	± 1.0 g	1
c. ESW Pump Facility	± 1.0 g	1
d. Ctmt. Structure	± 2.0 g	1
e. Auxiliary Bldg. SI Pump Suctions	± 1.0 g	1
f. SGB Piping	± 2.0 g	1
g. SGC Support	± 1.0 g	1
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)		
a. Ctmt. Base Slab	± 1.0 g	1
b. Ctmt. Oper. Floor	± 1.0 g	1
c. Reactor Support	± 1.0 g	1
d. Aux. Bldg. Base Slab	± 1.0 g	1
e. Aux. Bldg. Control Room Air Filters	± 1.0 g	1
f. Free Field	± 0.5 g	1
3. Triaxial Response-Spectrum Recorder (Passive)		
a. Ctmt. Base Slab	± 1.0 g	1

4. Triaxial Seismic Switches	ACCELERATION LEVEL			
	North	East	Vertical	
a. OBE Ctmt. Base Slab	0.09 g	0.09 g	0.13 g	1
b. SSE Ctmt. Base Slab	0.13 g	0.14 g	0.20 g	1
c. OBE Ctmt. Oper. Fl.	0.10 g	0.10 g	0.13 g	1
d. SSE Ctmt. Oper. Fl.	0.14 g	0.16 g	0.21 g	1
e. System Trigger	0.01 g	0.01 g	0.01 g	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. 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 Air 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 Annunciators for contact closure output in the Control Room shall be performed at least once per 184 days.

** The Bistable Trip Setpoint need not be determined during the performance of an ANALOG CHANNEL OPERATIONAL TEST.

TABLE 16.3-3

METEOROLOGICAL MONITORING INSTRUMENTATION

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

TABLE 16.3-4

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

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 Nobel Gas Monitor (GT-RIC-21B)	1	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

16.4 (3/4.4) REACTOR COOLANT SYSTEM

16.4.1 (3/4.4.2) SAFETY VALVES

16.4.1.1 (3.4.2.1) LIMITING CONDITION FOR OPERATION

SHUTDOWN

A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ±1% (The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.)

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.

16.4.1.1.1 (4.4.2.1) SURVEILLANCE REQUIREMENTS

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

16.4.1.1.2 BASES

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 Cold Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

16.4.2 (3/4.4.5) STEAM GENERATORS

16.4.2.1 (3.4.5) LIMITING CONDITION FOR OPERATION

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.

16.4.2.1.1 SURVEILLANCE REQUIREMENTS

16.4.2.1.1.a
(4.4.5.0)

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 Steam Generator Sample Selection and Inspection
(4.4.5.1)

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 Steam Generator Tube Sample Selection and Inspection
(4.4.5.2)

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 and the inspected tubes shall be verified acceptable per the acceptance criteria of Section 16.4.2.1.1.e. 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:

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 16.4.2.1.1.e.(a).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.

16.4.2.1.1.d Inspection Frequencies
(4.4.5.3)

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 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); 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 tube 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 <u>Acceptance Criteria</u>
(4.4.5.4)

(a) As used in this specification:

1) <u>Imperfection</u> 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) <u>Degradation</u> means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

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

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

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

6) <u>Plugging Limit</u> means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 48% of the nominal tube wall thickness;

7) <u>Unserviceable</u> 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), 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

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 inspection.

(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.

16.4.2.1.1.f Reports
(4.4.5.5)

(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.

16.4.2.1.2 BASES

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) a seismic occurrence greater than the Operating Basis Earthquake; and (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 48% 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. Results from WCAP-10043 have been used to establish the plugging limit.

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.

16.4.3 (3/4.4.7) CHEMISTRY

16.4.3.1 (3.4.7) LIMITING CONDITION FOR OPERATION

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.

16.4.3.1.1 (4.4.7) SURVEILLANCE REQUIREMENTS

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.

16.4.3.1.2 BASES

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits 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.

16.4.4 PRESSURE/TEMPERATURE LIMITS
(3/4.4.9)

16.4.4.1 LIMITING CONDITION FOR OPERATION
(3.4.9.2)

PRESSURIZER

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.

16.4.4.1.1 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.

16.4.4.1.2 BASES

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 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.

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.

See also Technical Specification Bases 3/4.4.9.

16.4.5 STRUCTURAL INTEGRITY
(3/4.4.10)

16.4.5.1 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 Section 16.4.5.1.1.

APPLICABILITY: ALL MODES.

ACTION:

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

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

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

d. The provisions of Technical Specification 3.0.4 are not applicable.

16.4.5.1.1 (4.4.10) SURVEILLANCE REQUIREMENTS

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).

16.4.5.1.2 BASES

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

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

16.4.6 (3/4.4.11) REACTOR COOLANT SYSTEM VENTS

16.4.6.1 (3.4.11) LIMITING CONDITION FOR OPERATION

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.

16.4.6.1.1 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. (This surveillance need not be performed on the untested reactor vessel head vent path until the first COLD SHUTDOWN to meet the OPERABILITY requirements.)

16.4.6.1.2 BASES

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 of a 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.

TABLE 16.4-1

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.

TABLE 16.4-2

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

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.

16.5 INTENTIONALLY BLANK

16.6 CONTAINMENT SYSTEMS
(3/4.6)

16.6.1 PRIMARY CONTAINMENT
(3/4.6.1)

16.6.1.1 LIMITING CONDITION FOR OPERATION
(3.6.1.2)

CONTAINMENT LEAKAGE

Containment leakage rates shall be limited to:

a. 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, 48.1 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, 48.1 psig.

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

ACTION:

a. With the overall integrated containment leakage rate exceeding 1.0 L_a, perform the ACTION of Technical Specification 3.6.1.1.

b. With the as left overall integrated containment leakage rate exceeding 0.75 L_a, restore the overall integrated leakage rate to less than 0.75 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a:

   1) Restore the combined leakage rate to less than 0.60 L_a within 4 hours, or

   2) Isolate each failed penetration within 4 hours by use of at least one closed manual valve or blind flange, or a deactivated automatic valve secured in the closed position, or

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

16.6.1.1.1 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 P_a, 48.1 psig, during each 10-year service period. (A one-time extension of the test interval is allowed for the third Type A test of the first 10-year service period, provided unit shutdown occurs no later than March 31, 1995 and performance of the Type A test occurs prior to unit restart following Refuel 7.)

b. If any periodic as found Type A test fails to meet L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive as found Type A tests fail to meet L_a, a Type A test shall be performed at least every 18 months until two consecutive as found Type A tests meet L_a, at which time the above test schedule may be resumed. The as left overall integrated containment leakage rate shall be less than 0.75 L_a;

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.

   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.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 48.1 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks, and
   2) Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Technical 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 Specifications 4.6.1.7.2 and 4.6.1.7.4, as applicable; and

g. The provisions of Technical Specification 4.0.2 are not applicable.

16.6.1.1.2 BASES

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident 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, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

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

The following exemptions have been granted to the requirements of Appendix J of 10 CFR Part 50:

1. Section III.A.1(a) - an exemption to the requirement to stop the Type A test if excessive leakage is determined. This exemption allows the satisfactory completion of the Type A test if the leakage can be isolated and appropriately factored into the results.

2. Section III.A.5(b) - an exemption for the acceptance criteria, in lieu of the present single criterion of the total measured containment leakage rate being less than 0.75 of the maximum allowable leakage rate, L_a, the "as found" allowable leakage rate will be L_a and the "as left" allowable leakage rate will be less than 0.75 L_a.

3. Section III.D.1(a) - an exemption that removes the requirement that the third test of each set of three Type A tests be conducted when the plant is shutdown for the 10-year plant inservice inspection.

Exemption 1 allows the continuance of a Type A test when excessive leakage is found provided that significant leaks are identified and isolated. After completion of the modified Type A test (i.e., a Type A test with the significant leakage paths isolated during the test), local leakage rates of those paths isolated during the modified Type A test will be measured before and after repairs to those paths. The adjusted "as found" leakage rate for the Type A test can be determined by adding the local leakage rates, measured before any repairs to those previously isolated leakage paths, to the containment integrated leakage determined in the modified Type A test, plus any leakage improvements (defined below) made prior to the test. This adjusted "as found" leakage rate is to be used in determining the scheduling of the periodic Type A test in accordance with Section III.A.6 of Appendix J.

The acceptability of the modified Type A test can be determined by calculating the adjusted "as left" containment overall integrated leakage rate and comparing it to the acceptance criteria of 0.75 L_a. The adjusted "as left" Type A leakage rate is determined by adding the local leakage rates, measured after any repairs and/or adjustments to those previously isolated leakage paths, to the leakage rate determined in the modified Type A test. It should be noted that additional adjustments for non-standard lineup and changes in containment volume are added to the measured leakage rate for both "as found" and "as left" determinations.

Leakage improvements are defined as the difference between the pre-repair LLRT and post-repair LLRT done on containment penetrations prior to the start of the Type A test.

The only differences between this approach and Appendix J requirements are that: (1) the potentially excessive leakage paths will be repaired and/or adjusted after the Type A test is completed; and (2) the Type A test leakage rate is partially determined by calculation rather than by direct measurement.

16.6.1.2 (3.6.1.6) <u>LIMITING CONDITION FOR OPERATION</u>

<u>CONTAINMENT VESSEL STRUCTURAL INTEGRITY</u>

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.

<u>APPLICABILITY</u>: MODES 1, 2, 3, and 4.

<u>ACTION</u>:

a. With the abnormal degradation indicated by the conditions in Section 16.6.1.2.1.a.(a).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, 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.

16.6.1.2.1 SURVEILLANCE REQUIREMENTS

16.6.1.2.1.a Containment Vessel Tendons
(4.6.1.6.1)

The structural integrity of the prestressing tendons of the containment vessel shall be demonstrated at the end of 1.5, 3.5 and 5.5 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:

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).

(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 determine 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 and one at mid-length) cut from each removed wire.

Failure to meet the requirements of Section 16.6.1.2.1.a.(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 exterior surface, that grease leakage that could affect containment integrity is not present, and

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) shall be considered as an indication of abnormal degradation of the containment structure.

16.6.1.2.1.b <u>End Anchorages and Adjacent Concrete Surfaces</u>
(4.6.1.6.2)

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.a. 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 <u>Containment Vessel Surfaces</u>
(4.6.1.6.3)

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.

16.6.1.2.2 BASES

This limitation ensures that the structural integrity of the containment vessel 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 48.1 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 tolerances 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

16.7.1.1 (3.7.2) LIMITING CONDITION FOR OPERATION

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 above requirements 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.

16.7.1.1.1 (4.7.2) SURVEILLANCE REQUIREMENTS

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.

16.7.1.1.2 BASES

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.

16.7.2 (3/4.7.8) SNUBBERS

16.7.2.1 (3.7.8) LIMITING CONDITION FOR OPERATION

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 16.7.2.1.1.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

16.7.2.1.1 SURVEILLANCE REQUIREMENTS
(4.7.8)

Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in lieu of the requirements of Technical Specification 4.0.5.

a. Inspection Types

As used here, 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 of 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 No. 67.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the 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; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Section 16.7.2.1.1.f. 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; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

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 16.7.2.1.1.f, 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. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Section 16.7.2.1.1.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new value of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 16.7-1. 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 that type of snubbers 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 group 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.

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. <u>Functional Test Acceptance Criteria</u>

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, 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.

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 16.7.2.1.1.e 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 Technical Specification 6.10.2.

16.7.2.1.2 BASES

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 On-Site 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 on the number of unacceptable snubbers found during the previous inspection in proportion to the size of the various snubber populations or categories. The Snubber Visual Inspection Interval is determined in accordance with Table 16.7-1. The maximum inspection interval can be as long as two refuel cycles but not more than 48 months, provided the requirements of Table 16.7-1 are met. A snubber is considered unacceptable if it fails the acceptance criteria of visual inspection.

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 the functional testing program, the visual scrutiny is 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 16.7-1, or

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

Figure 16.7-1 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.

16.7.3 (3/4.7.9) SEALED SOURCE CONTAMINATION

16.7.3.1 (3.7.9) LIMITING CONDITION FOR OPERATION

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.

16.7.3.1.1 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.

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.

16.7.3.1.2 <u>BASES</u>

The limitation 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 <u>AREA TEMPERATURE MONITORING</u>
(3/4.7.12)

16.7.4.1 <u>LIMITING CONDITION FOR OPERATION</u>
(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 (25°F for Electrical Penetration Rooms A and B).

<u>APPLICABILITY</u>: 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 (25°F for Electrical Penetration Rooms A and B), 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.

16.7.4.1.1 SURVEILLANCE REQUIREMENTS
(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.

16.7.4.1.2 BASES

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, except for Electrical Penetration Rooms A and B. These rooms have an alarm at ≤ 103°F with a maximum room temperature of 106°F.

TABLE 16.7-1

SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS Column A Extend Interval (Notes 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 population or category size 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, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or 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 B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

TABLE 16.7-1 (continued)

SNUBBER VISUAL INSPECTION INTERVAL (continued)

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 Columns 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.

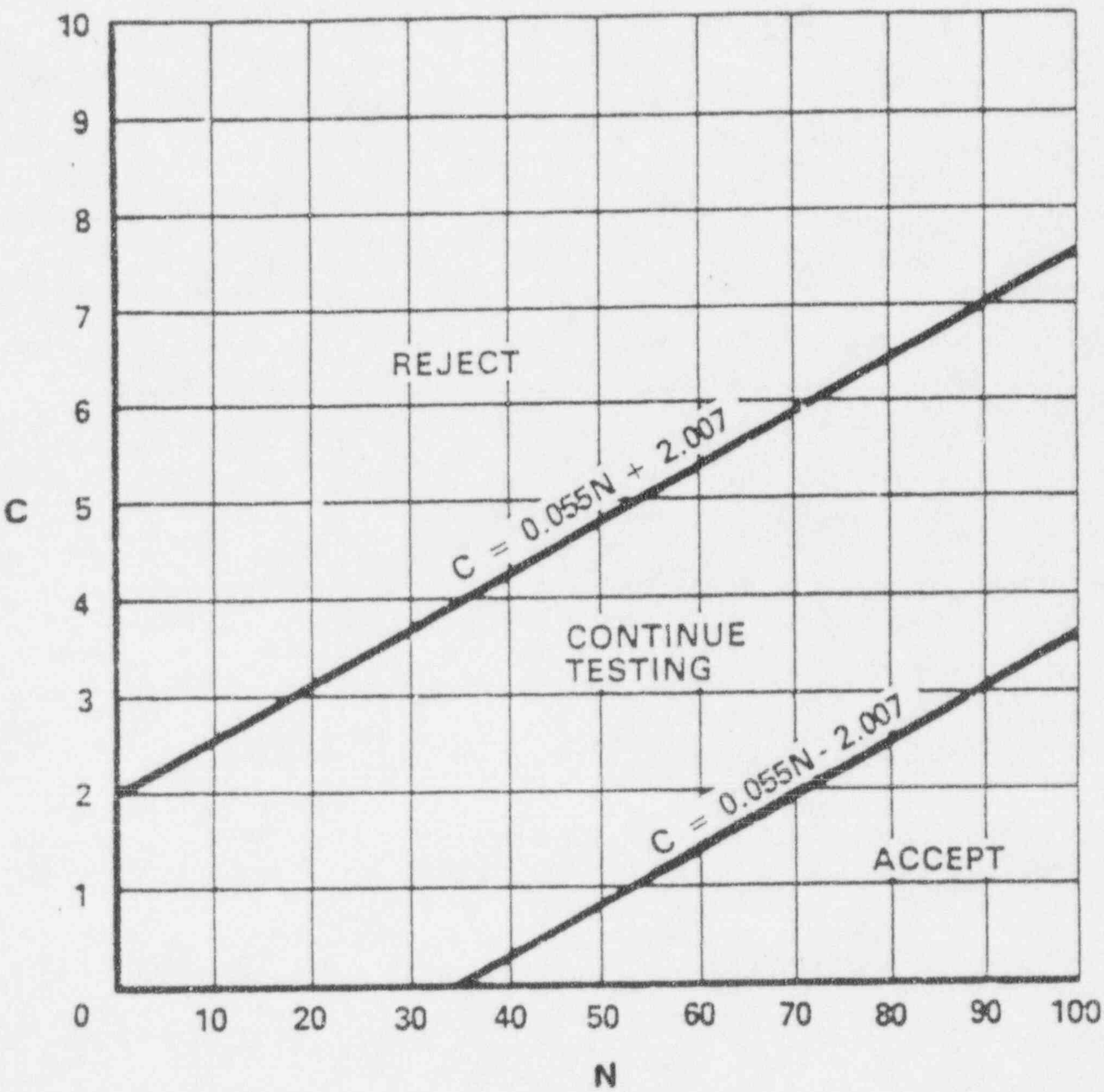

FIGURE 16.7-1

SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

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	106
17.	Electrical Penetration Room B	106
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

16.8.1.1 (3.8.4.1) LIMITING CONDITION FOR OPERATION

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

All containment penetration conductor overcurrent protective devices whose circuit limiting fault current exceeds the penetration rating 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 COLD SHUTDOWN within the following 30 hours.

16.8.1.1.1 (4.8.4.1) SURVEILLANCE REQUIREMENTS

All containment penetration conductor overcurrent protective devices whose circuit limiting fault current exceeds the penetration rating 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

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.

16.8.1.1.2 <u>BASES</u>

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, shall be 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 shall be made in accordance with Section 50.59 of 10 CFR Part 50.

16.9 REFUELING OPERATIONS
(3/4.9)

16.9.1 COMMUNICATIONS
(3/4.9.5)

16.9.1.1 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.

16.9.1.1.1 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.

16.9.1.1.2 BASES

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 REFUELING MACHINE
(3/4.9.6)

16.9.2.1 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 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 operations.

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.

16.9.2.1.1 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 to removal of the reactor vessel head 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 16.9.2.1.a.2) and by demonstrating an automatic load reduction trip when the load reduction exceeds the Setpoint of Section 16.9.2.1.a.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 removal of the reactor vessel head by performing a load test of at least 1250 pounds.

16.9.2.1.2 BASES

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.

16.9.3 (3/4.9.7) CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

16.9.3.1 (3.9.7) LIMITING CONDITION FOR OPERATION

Loads in excess of 2250 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage facility, except for the spent fuel pool transfer gates which may be moved over fuel assemblies in the spent fuel pool for refueling activities, fuel handling system maintenance, and transfer gate seal replacement.

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

ACTION:

a. With the above requirements 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.

16.9.3.1.1 (4.9.7) SURVEILLANCE REQUIREMENTS

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.

16.9.3.1.2 BASES

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.

The spent fuel pool transfer gates are excluded from this restriction because with a limited gate lift height, the spent fuel pool racks will absorb the impact of a dropped gate without damage to fuel assemblies. In addition, redundant trolleys and supports are used when moving the gates to preclude dropping a gate on the spent fuel racks, the time and distance the gates are moved over fuel is minimized as much as practical, and gate travel over fuel assemblies containing RCCAs is prohibited. The spent fuel pool transfer gates are only moved for refueling activities, fuel handling system maintenance, and to change gate seals.

16.9.4 WATER LEVEL - REACTOR VESSEL
(3/4.9.10)

16.9.4.1 LIMITING CONDITION FOR OPERATION
(3.9.10.2)

CONTROL RODS

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

16.9.4.1.1 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.

16.9.4.1.2 BASES

See Technical Specification Bases 3/4.9.10.

16.10 (3/4.10) SPECIAL TEST EXCEPTIONS

16.10.1 (3/4.10.5) POSITION INDICATION SYSTEM - SHUTDOWN

16.10.1.1 (3.10.5) LIMITING CONDITION FOR OPERATION

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.

16.10.1.1.1 (4.10.5) SURVEILLANCE REQUIREMENTS

The above required Position Indication System 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.

16.10.1.1.2 BASES

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

16.11.1.1 (3.11.1.4) LIMITING CONDITION FOR OPERATION

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 the liner being used to solidify radioactive waste.

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 Technical Specification 6.9.1.7.

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

16.11.1.1.1 (4.11.1.4) SURVEILLANCE REQUIREMENTS

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.

16.11.1.1.2 BASES

The tanks listed above 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

16.11.2.1 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 limit 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 on 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.

16.11.2.1.1 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 Section 16.3.1.6.

16.11.2.1.2 BASES

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.

16.11.3.1 (3.11.2.6) LIMITING CONDITION FOR OPERATION

GAS STORAGE TANKS

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 Technical Specification 6.9.1.7.

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

16.11.3.1.1 (4.11.2.6) SURVEILLANCE REQUIREMENTS

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.

16.11.3.1.2 BASES

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.