Sheet 11.1. Although not designed with the same high range, further diversity is available from the containment atmosphere radiation monitors (GT-RE-0031 and -0032) which display at the digital radiation monitoring panel SP067. Section 22 of NUREG-0830 specifically accepted the response to NUREG-0737 Item II.F.1 Attachment 3. Additional discussion is found in FSAR Section 18.2.12. For RVLIS, diversity is provided by the 46 core exit thermocouples, pressurizer level indication (BB-LI-0459A, -0460A, and -0461), and RCS subcooling monitor indication (BB-TI-1390A and B). Additional discussion is found in FSAR Table 7A-3 Data Sheet 1.4. If these alternate methods are used, new Action c does not require a plant shutdown, rather a Special Report is submitted within 14 days per Specification 6.9.2. The report provided to the NRC would discuss the preplanned alternate methods used, outline the cause of the inoperability, and provide a schedule for restoring the normal PAM channels.

New Action d applies when two hydrogen monitor channels are inoperable, requiring the restoration of one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour completion time is reasonable based on the backup capability of the Post Accident Sampling System to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA (which would cause core damage) would occur during this time. Consistent with the new STS, NUREG-1431, LCO 3.6.4.1 is deleted since LCO 3.3.3.6 contains the appropriate actions and surveillances.

~~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.~~

Determination of No Unreviewed Safety Question

The proposed changes to the Technical Specifications do not involve an unreviewed safety question because the operation of Callaway Plant in accordance with these proposed changes would not:

(1) Involve an increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. Overall protection system performance will remain within the bounds of the accident analyses documented in FSAR Chapter 15, WCAP-10961-P, and WCAP-11883 since

ATTACHMENT THREE

REPLACEMENT PAGES FOR ATTACHMENT 4 TO ULNRC-3023

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

TSI 64

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.

listed in the Bases of

e. ~~d.~~ The containment ~~leakage rates are within the limits of~~ Specification 3.6.1.~~0~~ 1, ~~and~~ and determined per Specification 4.6.1.1.d and are

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

f. Structural integrity is assured via the program described in Specification 6.8.5.c.

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."

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% Δk/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, the rod drop time of the individual full-length shutdown and control rods from the fully withdrawn position shall be demonstrated to be less than or equal to 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry 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.

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 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.

~~INSERT 4B →~~ (VOID)

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

(VOID)

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.

(RETAIN AS IS)

~~delete~~ (stet)

REVISION 1

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:

(RETAIN AS IS) ~~delete~~ (stet)

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.

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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.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

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

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

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

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

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

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 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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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.

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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.

~~delete~~ (stet)

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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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

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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. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a;

1.

INSERT 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.

INSERT A

d. By performing containment leakage rate testing, except for containment air locks, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions; and

e. By verifying containment structural integrity in accordance with the Containment Tendon Surveillance Program of Specification 6.8.5.c.

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. 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.

REVISION 1

~~3/4.4.5 STEAM GENERATORS~~ (RETAIN AS IS)

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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~~

~~STEAM GENERATORS (Continued)~~ (RETAIN AS IS)

~~DELETED~~ (stet)

~~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.

PROCEDURES AND PROGRAMS (Continued)

~~g.~~ 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 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.

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

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

The program shall include:

1. The limits for concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM and a surveillance program to ensure the limits are maintained.

2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of $\geq$0.5 rem to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY in the event of an uncontrolled release of the tanks' contents, consistent with Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases due to Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3. A surveillance program to ensure that the quantity of radioactivity contained in the following outdoor liquid radwaste tanks, that are not surrounded by liners, dikes, or walls capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste system, is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20.1 - 20.602, Appendix B (redesignated at 56FR23391, May 21, 1991) at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA, in the event of an uncontrolled release of the tanks' contents:

   a. Reactor Makeup Water Storage Tank,
   b. Refueling Water Storage Tank,
   c. Condensate Storage Tank, and

d. Outside temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

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

b. Reactor Coolant Pump Flywheel Inspection Program

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

c. Containment Tendon Surveillance Program

This program provides controls for monitoring tendon performance, including the effectiveness of the tendon corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial plant operation as well as periodic testing thereafter. The Containment Tendon Surveillance Program, and its inspection frequencies and acceptance criteria, shall be in accordance with the Callaway position on proposed Revision 3 of Regulatory Guide 1.35 dated April 1979.

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

TECHNICAL SPECIFICATION RETYPED PAGES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION	PAGE

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.

e. The containment leakage rates are determined per Specification 4.6.1.1.d and are within the limits listed in the Bases of Specification 3.6.1.1, and

f. Structural integrity is assured via the program described in Specification 6.8.5.c.

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."

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% Δk/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 hours the High Neutron Flux Trip Setpoint is reduced to less than or equa 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, the rod drop time of the individual full-length shutdown and control rods from the fully withdrawn position shall be demonstrated to be less than or equal to 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry 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.

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;

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 4.6.1.1.d for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a;

d. By performing containment leakage rate testing, except for containment air locks, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions; and

e. By verifying containment structural integrity in accordance with the Containment Tendon Surveillance Program of Specification 6.8.5.c.

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

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 4.6.1.1.d 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.

PROCEDURES AND PROGRAMS (Continued)

f. Radiological Environmental Monitoring Program (Continued)

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

6.8.5 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

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

The program shall include:

1. The limits for concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM and a surveillance program to ensure the limits are maintained.

2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY in the event of an uncontrolled release of the tanks' contents, consistent with Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases due to Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3. A surveillance program to ensure that the quantity of radioactivity contained in the following outdoor liquid radwaste tanks, that are not surrounded by liners, dikes, or walls capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste system, is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20.1 - 20.602, Appendix B (redesignated at 56FR23391, May 21, 1991) at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA, in the event of an uncontrolled release of the tanks' contents:

   a. Reactor Makeup Water Storage Tank,
   b. Refueling Water Storage Tank,

c. Condensate Storage Tank, and
d. Outside temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

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

b. Reactor Coolant Pump Flywheel Inspection Program

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

c. Containment Tendon Surveillance Program

This program provides controls for monitoring tendon performance, including the effectiveness of the tendon corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial plant operation as well as periodic testing thereafter. The Containment Tendon Surveillance Program, and its inspection frequencies and acceptance criteria, shall be in accordance with the Callaway position on proposed Revision 3 of Regulatory Guide 1.35 dated April 1979.

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

6.9 REPORTING REQUIREMENTS

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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ATTACHMENT FOUR

REPLACEMENT PAGES FOR ATTACHMENT 5 TO ULNRC-3023

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~~ Retain	~~Relocate~~ Retain	~~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 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, 13
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 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~~ 6
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~~ 6

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 Position Indicator
PORV Block Valve Position Indicator

~~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 and Storage Tank Radioactivity 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.~~ Deleted

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

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

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 3.6.1.2 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~~ add a reference to ~~a~~ new Specification 4.6.1.1.d that invokes 10CFR50, Appendix J. ~~that was relocated and instead reference corresponding PSAR Section 16.6.1.1.~~

13. ~~This~~ Specification 3.6.1.6 will be relocated and a Containment Tendon Surveillance Program ~~statement~~ will be incorporated into new Section 6.8.5. New SR 4.6.1.1.e will be added to implement this program.

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.~~ new SR 4.6.1.1.d.

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.~~ Deleted

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

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.~~ Deleted

REACTOR COOLANT SYSTEM

3.4.2.1	SAFETY VALVES - SHUTDOWN
3.4.5	STEAM GENERATORS (to be retained)
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

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.

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 will be relocated. ~~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 position indicator, and PORV block valve position indicator).

~~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 3.3.3.10 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~~ description 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).~~

TS3L

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. Rather than relocate this TS and add a program description to Section 6, this TS will be retained as 3/4.4.5.

(4) CONCLUSION

X 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 3.4.10 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~~ description will be added to new TS Section 6.8.5).

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 Pa and La 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 Pa and La 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 to reference new SR 4.6.1.1.d.

A new SR 4.6.1.1.d will be added that invokes 10CFR50, Appendix J.

TS6B

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~~ description will be added to new TS Section 6.8.5).

TS6F

New SR 4.6.1.1.e will be added to implement this program.

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 3.7.12 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 3.11.1.4 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~~ description 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~~ description 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~~ description will be added to new TS Section 6.8.5).