3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, meet the DNB design criteria during all normal operations and anticipated transients. In MODES 1 and 2 with less than all coolant loops in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal for removing decay heat; but, single failure considerations require all loops be in operation whenever the rod control system is energized and at least one loop be in operation when the rod control system is deenergized.

In MODE 4, a single reactor coolant loop or RHR loop provides sufficient heat removal for removing decay heat; but, single failure considerations require that at least 2 loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two RHR loops be OPERABLE.

In MODE 5, single failure considerations require that two RHR loops be OPERABLE. For support systems: Service Water (SW) and Component Cooling (CC), component redundancy is necessary to ensure no single active component failure will cause the loss of Decay Heat Removal. One piping path of SW and CC is adequate when it supports both RHR loops. The support systems needed before entering into the desired configuration (e.g., one service water loop out for maintenance in Modes 5 and 6) are controlled by procedures, and include the following:

- A requirement that two RHR, two CC and two SW pumps, powered from two different vital buses be kept operable
- A listing of the active (air/motor operated) valves in the affected flow path to be locked open or disabled

Note that four filled reactor coolant loops, with at least two steam generators with at least their secondary side water level greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop. This ensures that a single failure does not cause a loss of decay heat removal.

The operation of one Reactor Coolant Pump or one RHR Pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during Boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with Boron concentration reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to the POPS enable temperature specified in the PTLR are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer (thereby providing a volume into which the primary coolant can expand, or (2) by restricting the starting of Reactor Coolant Pumps to those times when secondary water temperature in each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 pounds per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperature. While in Mode 5 the safety valve requirement may be met by establishing a vent path of equivalent relieving capacity when no code safety valves are OPERABLE.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

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

Surveillance testing allows a ± 3% lift setpoint tolerance. However, to allow for drift during subsequent operation, the valves must be reset to within ± 1% of the lift setpoint following testing.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control RCS pressure and establish natural circulation.

3/4.4.5 RELIEF VALVES

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown.

3/4.4.5 RELIEF VALVES (continued)

B. Automatic control of PORVs to control reactor coolant system pressure. This is a function that reduces challenges to the code safety valves for overpressurization events, including an inadvertent actuation of the Safety Injection System.

C. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

D. Manual control of the block valve to : (1) unblock an isolated PORV to allow it to be used for manual and automatic control of Reactor Coolant System pressure (Items A & B), and (2) isolate a PORV with excessive seat leakage (Item C).

E. Manual control of a block valve to isolate a stuck-open PORV.

3/4.4.6 STEAM GENERATOR (SG) TUBE INTEGRITY

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.i, “Steam Generator (SG) Program,” and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that significantly affect burst or collapse. In that context, the term "significant" is defined as, "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB and draft Reg. Guide 1.121.

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, other than a steam generator tube rupture (SGTR), is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm per SG. The accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.7.2, "Operational Leakage," and limits primary-to-secondary leakage through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

The ACTION requirements are modified by a Note clarifying that the Actions may be entered independently for each SG tube. This is acceptable because the Action requirements provide appropriate compensatory actions for each affected SG tube. Complying with the Action requirements may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Action requirements.

If it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program, an evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging criteria define limits on SG tube that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. An action time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity. If the evaluation determines that the affected tube(s) have tube integrity, plant operation is allowed to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering HOT SHUTDOWN following the next refueling outage or SG inspection. This allowed outage time is acceptable since operation until the next inspection is supported by the operational assessment.

If SG tube integrity is not being maintained or the Action requirements are not met, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. The action times are reasonable based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

During shutdown periods the SGs are inspected as required by surveillance requirements and the Steam Generator Program. NEI 97-06, "Steam Generator Program Guidelines," and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period. The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities and inspection locations. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines.

The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.i contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 6.8.4.i until subsequent inspections support extending the inspection interval.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 6.8.4.i are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in size measurement and future growth. In addition, the tube plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). NEI 97-06 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria. The Frequency of prior to entering HOT SHUTDOWN following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

3/4.4.7 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.7.1 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.

3/4.4.7.2 OPERATIONAL LEAKAGE

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 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.

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.

Primary to Secondary Leakage Through Any One SG

The primary-to-secondary leakage rate limit applies to leakage through any one Steam Generator. The limit of 150 gallons per day per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, “The RCS operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day.” The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures. The dosage contribution from the tube leakage will be within 10 CFR 50.67 limits in the event of either a steam generator tube rupture or steam line break. The analyses are based on the total primary to secondary leakage from all SGs of 1 gallon per minute as a result of accident induced conditions.

Reactor Coolant System (RCS) Pressure Isolation Valves (PIV)

The function of the RCS PIVs is to separate the high pressure RCS from the attached low pressure systems. The PIV leakage limit applies to each individual valve listed in the Technical Requirements Manual. Leakage through both series PIVs in a line must be included as part of the IDENTIFIED LEAKAGE, governed by LCO 3.4.7.2, "Operational Leakage." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 4.4.7.2.1.d). A known component of the IDENTIFIED LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not

3/4.4.7.2 OPERATIONAL LEAKAGE (Continued)

RCS operational leakage if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

Actions

Unidentified leakage or identified leakage in excess of the LCO limits must be reduced to within limits within 4 hours. This action time allows time to verify leakage rates and either identify unidentified leakage or reduce leakage to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the reactor coolant pressure boundary (RCPB). If any pressure boundary leakage exists, or primary-to-secondary leakage is not within limit, or if unidentified or identified leakage cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. The reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary. The action times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Surveillances

Verifying RCS leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. Unidentified leakage and identified leakage are determined by performance of an RCS water inventory balance. The RCS water inventory must be met with the reactor at steady state operating conditions. The surveillance is modified by a Note that the surveillance is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

3/4.4.7.2 OPERATIONAL LEAKAGE (Continued)

Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If SR 4.4.7.2.1.c is not met, compliance with LCO 3.4.6, “Steam Generator Tube Integrity,” should be evaluated. The 150 gallons per day limit is measured at room temperature (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines). If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one Steam Generator. The Surveillance is modified by a Note that states that the surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary-to-secondary leakage determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines).

3/4.4.8

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3/4.4.9 SPECIFIC ACTIVITY

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a steam line break (SLB) or steam generator tube rupture (SGTR) to within the acceptance criteria. The iodine specific activity in the reactor coolant is limited to 1.0 µCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 600 µCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate acceptance criteria. The SLB and SGTR accident analyses show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the acceptance criteria.

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is $\leq$ 60.0 µCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend. The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note provided for both specific activities permits the use of the provisions of Technical Specification 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Actions 3.4.9.a while the DOSE EQUIVALENT I-131 LCO limit is not met and Action 3.4.9.b while the DOSE EQUIVALENT XE-133 is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

If the Action and associated completion time of LCO 3.4.9.a.1 is not met or if the DOSE EQUIVALENT I-131 is > 60 µCi/gm, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

If the Action and the associated completion time of LCO 3.4.9.b is not met the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

3/4.4.9 SPECIFIC ACTIVITY (Continued)

Surveillance Requirement 4.4.9.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant in accordance with the Surveillance Frequency Control Program (SFCP). This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in the noble gas specific activity. Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Frequency within the SFCP considers the low probability of a gross fuel failure during this time. Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the Surveillance calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

Surveillance Requirement 4.4.9.2 is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The Frequency identified in the SFCP is adequate to trend changes in the iodine activity level, considering noble gas activity is routinely monitored. The Frequency, between 2 and 6 hours after a power change $\geq$ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

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

1) The reactor coolant temperature and pressure and system heatup and cooldown rate (with the exception of the pressurizer) shall be limited in accordance with those specified in the PTLR for the service period specified therein.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines specified in the PTLR. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b) The limits specified in the PTLR assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) The limit lines specified in the PTLR shall be calculated periodically using methods provided in Technical Specification 6.9.1.11.

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

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

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

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. In order to provide sufficient safety margins for protection against non-ductile failure, the pressure/temperature (P/T) limit curves for heatup and cooldown, inservice leak and hydrostatic testing, and criticality are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the 1/4T and 3/4T locations of the beltline and extended beltline materials for the applicability term stated in the PTLR. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in UFSAR Section 5.2. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. To ensure that the radiation embrittlement effects are accounted for in the calculation of the P/T limit curves, the most limiting RT_{NDT} includes the radiation induced shift, ΔRT_{NDT} corresponding to the end of the period for which curves are generated. This adjusted reference temperature (ART) can be predicted based upon the fluence and the copper and nickel content of the material in question. The ART is based upon the largest value of RT_{NDT} computed by the methodology presented in Regulatory Guide 1.99, Revision 2. The PTLR contains the results of the ART evaluations. The predicted neutron fluence, as a function of Effective Full Power Years (EFPY), has been calculated based on the guidance of Regulatory Guide 1.190. The PTLR contains the results of the fluence evaluations.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

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

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The heatup and cooldown curves are composite curves, constructed based on a point-by-point comparison of the steady-state and finite rate data, as well as the reactor vessel and head flange, to identify the most restrictive curve. The use of the composite curve is necessary to set conservative heatup/cooldown limitations because at any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations have more restrictive limits, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve requirement is that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for inservice leak rate testing.

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

The OPERABILITY of two POPSs valves or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to the POPS enable temperature specified in the PTLR. Either POPS valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of an Intermediate Head Safety Injection pump and its injection into a water solid RCS, or the start of a High Head Safety Injection pump in conjunction with a running Positive Displacement pump and its injection into a water solid RCS. The minimum electrical power sources required to assure POPS operability (based on POPS meeting the single failure criteria) consist of a normal (via offsite power) and an emergency (via batteries) power source for each train of POPS. Emergency diesel generators are not required for POPS to meet single failure criteria and therefore are not required for POPS OPERABILITY.

LCO 3.0.4.b is not applicable to an inoperable POPS when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable POPS. The provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

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