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CHAPTER B 3.4

REACTOR COOLANT SYSTEM

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR limits (Ref. 2). Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution limits are satisfied per LCO 3.1.4, "Rod Group Alignment Limits;" LCO 3.1.5, "Shutdown Bank Insertion Limits;" LCO 3.1.6, "Control Bank Insertion

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of 2195 psig and the RCS average temperature limit of 590.1°F, as specified in the COLR, correspond to analytical limits of 2175 psig and 592.7°F used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. The limit values for pressurizer pressure and RCS average temperature are specified in the COLR to provide operating and analysis flexibility from cycle to cycle. Operating within these limits will result in meeting the DNBR limits in the event of a DNB limited transient.

The RCS total flow rate limit contains a measurement error of 2.1% based on performing a precision heat balance and using the result to normalize the RCS flow rate indicators. Potential fouling of the feedwater flow venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a bias of 0.1% for undetected fouling of the feedwater flow venturi raises the nominal flow measurement allowance to 2.2%. The RTDP thermal-hydraulic analyses assume a total RCS minimum measured flow of 382,630 gpm, which is 2.2% greater than the thermal design flow of 374,400 gpm used in non-RTDP analyses.

Any fouling that might bias the flow rate measurement greater than the 0.1% bias for undetected fouling of the feedwater flow venturi can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The numerical values for pressure, temperature, and flow rate have been adjusted for instrument error as discussed above.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In

(continued)

BASES

APPLICABILITY (continued)

all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients, namely a THERMAL POWER ramp increase or decrease > 5% RTP per minute or a THERMAL POWER step increase or decrease > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive.

The DNBR limits are provided in SL 2.1.1, "Reactor Core SLs" (Ref. 2). The conditions which define the DNBR limits are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and reduce the potential for violation of the accident analysis limits.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

(continued)

BASES (Continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Periodic verification that pressurizer pressure is greater than or equal to the limit specified in the COLR ensures that pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. This infomation is used to assess potential degradation and to verify operation is within safety analysis assumptions. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.2

Periodic verification that RCS average temperature is less than or equal to the limit specified in the COLR ensures that temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. This information is used to assess potential degradation and to verify operation is within safety analysis assumptions. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.3

Periodic verification that RCS total flow rate is greater than or equal to 382,630 gpm is performed using the installed flow intrumentation. This information is used to assess potential degradation and to verify operation is within safety analysis assumptions. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance allows the installed RCS flow instrumentation to be normalized and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. When performing a precision heat balance, the instrumentation used for determining steam pressure, feedwater temperature, and feedwater venturi Δp in the calorimetric calculations shall be calibrated within 7 days prior to performing the heat balance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.4 (continued)

The Frequency of once after each refueling prior to THERMAL POWER exceeding 75% RTP, and in accordance with the Surveillance Frequency Control Program, reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows this SR to be performed by RCS flow calibration after each refueling prior to THERMAL POWER exceeding 75% RTP. The determination of RCS flow through a precision heat balance is not required prior to entry into MODE 1 because RCS flow indication is available (RCS flow meters), RCS flow is calculated prior to exceeding 75% RTP and the surveillance from the previous cycle is still current. The precision heat balance flow rate surveillance is typically performed after reaching full power following a refueling outage and suitable plant conditions are established.

REFERENCES

1. FSAR, Chapter 15.
2. SL 2.1.1, "Reactor Core Safety Limits (SLs)."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil ductility transition reference temperature when the reactor is critical.

APPLICABLE SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 557°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) with an operating loop's temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

The special test exception of LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2," permits PHYSICS TESTS to be performed at ≤ 5% RTP with operating RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow operating RCS loop average temperatures to fall below $T_{no\ load}$, which may cause operating RCS loop average temperatures to fall below the temperature limit of this LCO.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with $k_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with $k_{eff} < 1.0$ in an orderly manner and without challenging plant systems.

(continued)

BASES (Continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above 551°F.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, or the pressurizer surge line, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3). These methods are discussed in detail in WCAP - 14894, approved by NRC for Callaway (Refs. 8 and 9).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility transition reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and

(continued)

BASES

BACKGROUND (continued)

Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. 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 are more restrictive, 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 includes the Reference 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA). analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

(continued)

BASES (Continued)

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer and the pressurizer surge line. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Exceeding the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer or the pressurizer surge line.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum

(continued)

BASES

APPLICABILITY (continued)

pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig within 36 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.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required when RCS pressure and temperature conditions are undergoing planned changes. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. WCAP-14040-NP-A.

2. 10 CFR 50, Appendix G.

3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

4. ASTM E 185-73.

(continued)

BASES

REFERENCES (continued)

5. 10 CFR 50, Appendix H.

6. Regulatory Guide 1.99, Revision 2, May 1988.

7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

8. WCAP - 14894, "Callaway Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," July 1997.

9. Amendment No. 124 to Facility Operating License NPF-30 dated April 2, 1998.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The reactor coolant is circulated through four loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

All of the accident/safety analyses performed at full rated thermal power assume that all four RCS loops are in operation as an initial condition. Some accident/safety analyses have been performed at zero power conditions assuming only two RCS loops are in operation to

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

conservatively bound lower modes of operation. The events which assume only two RCPs in operation include the uncontrolled RCCA bank withdrawal from subcritical and the hot zero power rod ejection events. While all accident/safety analyses performed at full rated thermal power assume that all the RCS loops are in operation, selected events examine the effects resulting from a loss of RCP operation. These include the complete and partial loss of forced RCS flow, RCP locked rotor, and RCP shaft break events. For each of these events, it is demonstrated that all the applicable safety criteria are satisfied. For the remaining accident/safety analyses, operation of all four RCS loops during the transient up to the time of reactor trip is assumed thereby ensuring that all the applicable acceptance criteria are satisfied. Those transients analyzed beyond the time of reactor trip were examined assuming that a loss of offsite power occurs which results in the RCPs coasting down.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the limit values, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

An OPERABLE RCS loop consists of an OPERABLE RCP and an OPERABLE SG. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow.

APPLICABILITY

In MODES 1 and 2, the reactor, when critical, has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

(continued)

BASES

APPLICABILITY (continued)

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification that each RCS loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

APPLICABLE SAFETY ANALYSES

Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the Rod Control System. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM.

Therefore, in MODE 3 with the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least two RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

The operation of one RCP in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity

(continued)

APPLICABLE SAFETY ANALYSES (continued)

changes during RCS boron concentration reductions. The reactivity change rate associated with boron reduction will, therefore, be within the transient mitigation capability of the Boron Dilution Mitigation System (BDMS). With no reactor coolant loop in operation in either MODES 3, 4, or 5, boron dilutions must be terminated and all dilution sources isolated. The boron dilution analysis in these MODES takes credit for the mixing volume associated with having at least one reactor coolant loop in operation. LCO 3.3.9, "Boron Dilution Mitigation System (BDMS)," contains the requirements for the BDMS.

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or 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.

RCS Loops - MODE 3 satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two RCS loops be OPERABLE. In MODE 3 with the Rod Control System capable of rod withdrawal, two RCS loops must be in operation. Two RCS loops are required to be in operation in MODE 3 with the Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

When the Rod Control System is not capable of rod withdrawal, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that redundancy for heat removal is maintained.

The Note permits all RCPs to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are required to be performed without flow or pump noise. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again.

(continued)

BASES

LCO (continued)

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Introduction of reactor makeup water into the RCS from the Chemical and Volume Control System mixing tee is not permitted, operation of CVCS resin vessels configured with resin for dilution during normal operation is not permitted, and operation of the purge line associated with flushing the CVCS letdown radiation monitor is not permitted when no RCS loop is in operation. Note the CVCS resin vessels include the resin vessels of its subsystem the BTRS. Boron dilution with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";

(continued)

BASES

APPLICABILITY (continued)

LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

B.1

If restoration is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If the required RCS loop is not in operation, and the Rod Control System is capable of rod withdrawal, the Required Action is either to restore the required RCS loop to operation or to place the Rod Control System in a condition incapable of rod withdrawal (e.g., by de-energizing all CRDMs, by opening the RTBs, or de-energizing the motor generator (MG) sets). When the Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the Rod Control System must be rendered incapable of rod withdrawal. The Completion Times of 1 hour to restore the required RCS loop to operation or defeat the Rod Control System is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

(continued)

BASES

ACTIONS (continued)

D.1, D.2, and D.3

If four RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, place the Rod Control System in a condition incapable of rod withdrawal (e.g., by de-energizing all CRDMs, by opening the RTBs, or de-energizing the MG sets). All operations involving introduction of coolant, into the RCS, with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and defeating the Rod Control System removes the possibility of an inadvertent rod withdrawal. Suspending the introduction of coolant, into the RCS, with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. Introduction of reactor makeup water into the RCS from the Chemical and Volume Control System mixing tee is not permitted, operation of CVCS resin vessels configured with resin for dilution during normal operation is not permitted, and operation of the purge line associated with flushing the CVCS letdown radiation monitor is not permitted when no RCS loop is in operation, consistent with Required Action C.1 of LCO 3.3.9, “Boron Dilution Mitigation System (BDMS).” The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification that the required loops are in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 7% for required RCS loops. If the SG secondary side narrow range

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.2 (continued)

water level is < 7%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.5.3

Verification that the required RCPs are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR Section 15.4.6.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event.

The operation of one RCP in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during RCS boron concentration reductions. The reactivity change rate associated with boron reduction will, therefore, be within the transient mitigation capability of the Boron Dilution Mitigation System (BDMS). With no reactor coolant loop in operation in either MODES 3, 4, or 5, boron dilutions must be terminated and all dilution sources isolated. The boron dilution analysis in these MODES takes credit for the mixing volume associated with having at least one reactor coolant loop in operation. LCO 3.3.9, "Boron Dilution Mitigation System (BDMS)," contains the requirements for the BDMS.

RCS Loops -MODE 4 satisfies Criterion 4 of 10CFR50.36(c)(2)(ii).

(continued)

BASES (Continued)

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise. The 1 hour time period is adequate to perform the necessary testing, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Introduction of reactor makeup water into the RCS from the Chemical and Volume Control System mixing tee is not permitted, operation of CVCS resin vessels configured with resin for dilution during normal operation is not permitted, and operation of the purge line associated with flushing the CVCS letdown radiation monitor is not permitted when no RCS loop is in operation. Note that CVCS resin vessels include the resin vessels of its subsystem the BTRS. Boron dilution with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ 275°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

(continued)

BASES

LCO (continued)

An OPERABLE RCS loop is comprised of an OPERABLE RCP and an OPERABLE SG, which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

Management of voids is important to RHR System OPERABILITY. The RHR System is OPERABLE when it is sufficiently filled with water to perform its specified safety function.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one required loop is inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

The unit must be brought to MODE 5 within 24 hours if, as indicated in the Note to Required Action A.2, one RHR loop is OPERABLE. Bringing the unit to MODE 5 is a conservative action with regard to decay heat

(continued)

ACTIONS

A.1 and A.2 (continued)

removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 (≤ 200°F) rather than MODE 4 (200 to 350°F). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

B.1 and B.2

If no loop is OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant, into the RCS, with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation from at least one RCP for proper mixing so that inadvertent criticality can be prevented. Suspending the introduction of coolant, into the RCS, with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. Introduction of reactor makeup water into the RCS from the Chemical and Volume Control System mixing tee is not permitted, operation of CVCS resin vessels configured with resin for dilution during normal operation is not permitted, and operation of the purge line associated with flushing the CVCS letdown radiation monitor is not permitted when no RCS loop is in operation, consistent with Required Action C.1 of LCO 3.3.9, “Boron Dilution Mitigation System (BDMS).” The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification that one RCS or RHR loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Surveillance Frequency is based on operating experience, equipment

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1 (continued)

reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 7% for required RCS loops. If the SG secondary side narrow range water level is < 7%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available the required pump. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.6.4

RHR System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the required RHR loop(s) and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

Selection of RHR System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walkdowns to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.4 (continued)

system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as standby versus operating conditions.

The RHR System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the RHR System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RHR System locations susceptible to gas accumulation are monitored, and if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

This SR is modified by a Note that states the SR is not required to be performed until 12 hours after entering MODE 4. In a rapid shutdown, there may be insufficient time to verify all susceptible locations prior to entering MODE 4.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

(continued)

BASES (Continued)

REFERENCES	1.	FSAR Section 15.4.6.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. When the RHR heat exchangers are used, heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The heat sink for the Component Cooling Water System is in turn normally provided by the Service Water System or Essential Service Water System, as determined by system availability.

While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification but is not sufficient for the boron dilution analysis discussed below.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or

(continued)

BASES

BACKGROUND (continued)	maintaining two SGs with secondary side wide range water levels above 86% to provide an alternate method for decay heat removal via natural circulation.
APPLICABLE SAFETY ANALYSES	In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The operation of one RCP in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during RCS boron concentration reductions. The reactivity change rate associated with boron reduction will, therefore, be within the transient mitigation capability of the Boron Dilution Mitigation System (BDMS). With no reactor coolant loop in operation in either MODES 3, 4, or 5, boron dilutions must be terminated and all dilution sources isolated. The boron dilution analysis in these MODES takes credit for the mixing volume associated with having at least one reactor coolant loop in operation. LCO 3.3.9, "Boron Dilution Mitigation System (BDMS)," contains the requirements for the BDMS. RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10CFR50.36(c)(2)(ii).
LCO	The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side wide range water level ≥ 86%. As shown in Reference 3, any narrow range level indication above 7% will ensure the SG tubes are covered. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. If both RHR loops are OPERABLE, either RHR loop may be the operating loop. If the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side wide range water levels ≥ 86%. Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE. Note 1 permits all RHR pumps to be removed from operation ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise. The 1 hour time period is

(continued)

BASES

LCO (continued)

adequate to perform the necessary testing, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Introduction of reactor makeup water into the RCS from the Chemical and Volume Control System mixing tee is not permitted, operation of CVCS resin vessels configured with resin for dilution during normal operation is not permitted, and operation of the purge line associated with flushing the CVCS letdown radiation monitor is not permitted when no RCS loop is in operation. Note that CVCS resin vessels include the resin vessels of its subsystem the BTRS. Boron dilution with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with any RCS cold leg temperature ≤ 275°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

(continued)

BASES

LCO (continued)

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. Electrical power source and distribution requirements for the RHR loops are as specified per LCO 3.8.2, “AC Sources - Shutdown”; LCO 3.8.5, “DC Sources - Shutdown”; LCO 3.8.8, “Inverters - Shutdown,” and LCO 3.8.10, “Distribution Systems - Shutdown,” consistent with the Bases for those Technical Specifications for reduced requirements during shutdown conditions, subject to the provisions and limitations described in the Bases.

Management of gas voids is important to RHR System OPERABILITY. The RHR System is OPERABLE when it is sufficiently filled with water to perform its specified safety function.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side wide range water level of at least two SGs is required to be ≥ 86%.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side wide range water levels < 86%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Notes 1 and 4, or if no loop is OPERABLE, all operations involving introduction of coolant, into the RCS, with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent inadvertent criticality during a boron dilution, forced circulation from at least one RCP is required to provide proper mixing. Suspending the introduction of coolant, into the RCS, with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. Introduction of reactor makeup water into the RCS from the Chemical and Volume Control System mixing tee is not permitted, operation of CVCS resin vessels configured with resin for dilution during normal operation is not permitted, and operation of the purge line associated with flushing the CVCS letdown radiation monitor is not permitted when no RCS loop is in operation, consistent with Required Action C.1 of LCO 3.3.9, “Boron Dilution Mitigation System (BDMS).” The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification that the required loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side wide range water levels are ≥ 86% ensures an alternate decay heat removal method is available via natural circulation in the event that the second RHR loop is not OPERABLE. As shown in Reference 3, any narrow range level indication above 7% will ensure the SG tubes are covered. If both RHR loops are OPERABLE, this Surveillance is not needed. The Surveillance Frequency is based on operating experience,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.7.2 (continued)

equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side wide range water level is ≥ 86% in at least two SGs, this Surveillance is not needed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.7.4

RHR System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the required RHR loop(s) and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

Selection of RHR System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walkdowns to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as standby versus operating conditions.

The RHR System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the RHR System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.7.4 (continued)

declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RHR System locations susceptible to gas accumulation are monitored, and if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of SGs to Remove Decay Heat by Natural Circulation."

2. FSAR Section 15.4.6.

3. TDB-001, "Tank Data Book, Steam Generators EBB01 (A,B,C,D)."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

During RHR system operation, heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The flow provided by one RHR loop is adequate for decay heat removal.

The operation of one RCP in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during RCS boron concentration reductions. The reactivity change rate associated with boron reduction will, therefore, be within the transient mitigation capability of the Boron Dilution Mitigation System (BDMS). With no reactor coolant loop in operation in either MODES 3, 4, or 5, boron dilutions must be terminated and all dilution sources isolated. The boron dilution analysis in these MODES takes credit for the mixing volume associated with having at least one reactor coolant loop in operation. LCO 3.3.9, "Boron Dilution Mitigation System (BDMS)," contains the requirements for the BDMS.

RCS loops in MODE 5 (loops not filled) satisfies Criterion 4 of 10CFR50.36(c)(2)(ii).

(continued)

BASES (Continued)

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

Note 1 permits all RHR pumps to be removed from operation for ≤ 1 hour. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained at least 10°F below saturation temperature. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained or draining operations when RHR forced flow is stopped. Introduction of reactor makeup water into the RCS from the Chemical and Volume Control System mixing tee is not permitted, operation of CVCS resin vessels configured with resin for dilution during normal operation is not permitted, and operation of the purge line associated with flushing the CVCS letdown radiation monitor is not permitted when no RCS loop is in operation. Note that CVCS resin vessels include the resin vessels of its subsystem the BTRS.

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

Note 3 clarifies that the Service Water System may serve as the heat sink for one of the two required RHR loops, provided that one train of the Essential Service Water System serves as the heat sink for the other RHR loop. This allowance is dependent upon the plant not being in a reduced-inventory, hot-core condition. A hot core is the present fuel cycle core within the reactor (before refueling). Reduced inventory corresponds to an RCS level lower than 3 feet below the reactor vessel flange with fuel in the reactor vessel. Since a reduced inventory, hot-core condition is an elevated risk condition for the plant during Mode 5, the restriction in the note for this condition ensures the decay heat removal function performed by the RHR system is supported by the safety-related Essential Service Water System (in lieu the non-safety Service Water System).

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

(continued)

BASES

LCO (Continued)

Electrical power source and distribution requirements for the RHR loops are as specified per LCO 3.8.2, "AC Sources - Shutdown"; LCO 3.8.5, "DC Sources - Shutdown"; LCO 3.8.8, "Inverters - Shutdown," and LCO 3.8.10, "Distribution Systems - Shutdown," consistent with the Bases for those Technical Specifications for reduced requirements during shutdown conditions, subject to the provisions and limitations described in the Bases.

Management of gas voids is important to RHR System OPERABILITY. The RHR System is OPERABLE when it is sufficiently filled with water to perform its specified safety function.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

Since LCO 3.4.8 contains Required Actions with immediate Completion Times, it is not permitted to enter LCO 3.4.8 from either LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled" or from MODE 6 unless the requirements of LCO 3.4.8 are met.

ACTIONS

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required RHR loops are OPERABLE or in operation, except during conditions permitted by Note 1, all operations involving introduction of

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

coolant, into the RCS, with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. Boron dilution requires forced circulation from at least one RCP for proper mixing so that inadvertent criticality can be prevented. Suspending the introduction of coolant, into the RCS, with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. Introduction of reactor makeup water into the RCS from the Chemical and Volume Control System mixing tee is not permitted, operation of CVCS resin vessels configured with resin for dilution during normal operation is not permitted, and operation of the purge line associated with flushing the CVCS letdown radiation monitor is not permitted when the RCS loops are not filled or when no RCS loop is in operation, consistent with Required Action C.1 of LCO 3.3.9, “Boron Dilution Mitigation System (BDMS).” The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification that one loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.8.2

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.8.3

RHR System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the RHR loops and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

Selection of RHR System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walkdowns to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as standby versus operating conditions.

The RHR System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the RHR System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RHR System locations susceptible to gas accumulation are monitored, and if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.8.3 (continued) The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.
REFERENCES	1. FSAR Section 15.4.6.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level and the required heaters. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat. Two groups of backup pressurizer heaters are normally powered via the Class 1E 4.16-kV buses. The heater loads will be shed after a safety injection or bus

(continued)

BASES

BACKGROUND (continued)	undervoltage signal and manually sequenced back onto the Class 1E 4.16-kV buses.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The presence of a steam bubble in the pressurizer, assured by the pressurizer level control program, satisfies Criterion 2 of 10CFR50.36(c)(2)(ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1657 cubic feet, which is equivalent to 92%, ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is consistent with analytical assumptions.

The LCO requires two groups of OPERABLE backup pressurizer heaters, each with a capacity ≥ 150 kW, capable of being powered from either the offsite power source or the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The backup pressurizer heaters may be controlled from either the main control board or the auxiliary shutdown panel.

(continued)

BASES (Continued)

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters, capable of being powered from either the offsite power source or the emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1, A.2, A.3, and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the LCO limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the LCO limit is the same as the Pressurizer Water Level - High Trip.

If the pressurizer water level is not within the limit, action must be taken to bring the plant to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3, with all rods fully inserted and incapable of withdrawal (e.g., by de-energizing all CRDMs, by opening the RTBs, or de-energizing the motor generator (MG) sets). Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

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.

B.1

If one required group of backup pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss

(continued)

BASES

ACTIONS

B.1 (continued)

of offsite power would be unlikely in this period. Pressure control may be maintained during this time using the remaining OPERABLE backup pressurizer heater group or the variable heater group.

C.1 and C.2

If one group of backup pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 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.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the LCO limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated backup pressurizer heaters are verified to have a capacity $\geq$ 150 kW (for each heater group). This is done by energizing the heaters and measuring circuit current. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 15.
2. NUREG-0737, November 1980.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Trip System, overpressure protection for the RCS. The pressurizer safety valves are of the pop type. The valves are spring loaded and self actuated by direct fluid pressure with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are self actuating, they are considered independent components. The minimum relief capacity for each valve, 420,000 lb/hr at 2485 psig plus 3% accumulation, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves which is divided equally between the three valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 with the reactor vessel head on; however, in MODE 4 with one or more RCS cold leg temperatures ≤ 275°F, MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."

The upper and lower pressure limits are based on the tolerance requirements assumed in the safety analyses. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

(continued)

BASES

BACKGROUND (continued)	The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal at full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load/turbine trip;

d. Loss of normal feedwater;

e. Loss of non-emergency AC power to station auxiliaries;

f. Locked rotor;

g. Feedwater line break; and

h. Rod cluster control assembly ejection.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation occurs in the FSAR Chapter 15 analysis of events c and f (above) and may be required for any of the above events to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

The three pressurizer safety valves are set to open at 2460 psig (slightly below the RCS design pressure of 2485 psig), and within the specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the tolerance requirements assumed in the safety analyses.

(continued)

BASES

LCO (continued)

The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the COMS arming temperature, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 with any RCS cold leg temperature ≤ 275°F, MODE 5, or MODE 6 with the reactor vessel head on because COMS is in service. Overpressure protection is not required in MODE 6 with the reactor vessel head removed (vent path ≥ 2 square inches).

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperature ≤ 275°F within 24 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. With any RCS cold leg temperatures at or below 275°F, overpressure protection is provided by the COMS. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

This SR requires testing specified for the pressurizer safety valves in the INSERVICE TESTING PROGRAM. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy this SR. No additional requirements are specified.

The pressurizer safety valve setpoint is ±2% for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

2. FSAR, Chapter 15.

3. WCAP-7769, Rev. 1, June 1972.

4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are safety-related DC solenoid operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive seat leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the block valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The power supplies to the PORVs and their block valves are Class 1E. The automatic pressure relief signal actuation circuitry, the manual controls, and the cold overpressure mitigation system (COMS) portion of the actuation circuitry are also Class 1E. The PORVs and their associated block valves are powered from two separate safety trains (Ref. 2).

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure -High reactor trip setpoint for all design transients up to and including the design step load decrease with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for cold

(continued)

BASES

BACKGROUND (continued)	overpressure mitigation. See LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."

APPLICABLE SAFETY ANALYSES

Plant operators may employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

For the inadvertent ECCS actuation at power event (a Condition II event), the safety analysis (Ref. 1) credits operator actions from the main control room to terminate flow from the normal charging pump (NCP) and to open both PORV block valves (assumed to initially be closed) to assure the availability of at least one PORV for automatic pressure relief. Analysis results indicate that water relief through the pressurizer safety valves, which could result in the Condition II event degrading into a Condition III event if the safety valves did not reseat, is precluded if operator actions are taken within the times assumed in the Reference 1 analysis to terminate NCP flow and to assure at least one PORV is available for automatic pressure relief. The assumed operator action times conservatively bound the times measured during simulator exercises. Therefore, automatic PORV operation is an assumed safety function in MODES 1, 2, and 3. The PORVs are equipped with automatic actuation circuitry and manual control capability. The PORVs are considered OPERABLE in either the automatic or manual mode, as long as the automatic actuation circuitry is OPERABLE and the PORVs can be made available for automatic pressure relief by timely operator actions (Ref. 1) to open the associated block valves (if closed) and assure the PORV handswitches are in the automatic operation position. The automatic mode is the preferred configuration, as this provides the required pressure relieving capability without reliance on operator actions.

The PORVs are also modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR), pressurizer volume, or hot leg saturation criteria are examined (Ref. 3). By assuming PORV actuation, the primary pressure remains below the high pressurizer pressure trip setpoint. The DNBR calculation is more conservative, the pressurizer water volume is maximized, and the hot leg saturation temperature is reduced for those transients assuming

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

PORV operation. Events that assume this condition include turbine trip, loss of normal feedwater, loss of non-emergency AC power to station auxiliaries, and feedline break (Ref. 3). Automatic operation is assumed in the Reference 3 analyses , but operation of the PORVs has a detrimental impact on the results of the analysis.

Pressurizer PORVs satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

The LCO also requires the PORVs and their automatic actuation circuitry to be OPERABLE, in conjunction with the capability to manually open their associated block valves and assure the availability of the PORVs for automatic pressure relief, to mitigate the effects associated with an inadvertent ECCS actuation at power event. The PORVs are considered OPERABLE in either the automatic or manual mode, as long as the automatic actuation circuitry is OPERABLE and the PORVs can be made available for automatic pressure relief by timely operator actions (Ref. 1) to open the associated block valves (if closed) and assure the PORV handswitches are in the automatic operation position. The automatic mode is the preferred configuration, as this provides the required pressure relieving capability without reliance on operator actions.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open and energized, or closed and energized, with the capability to be cycled, since the required safety functions of the block valve are accomplished by manual operation to cycle the block valve. Although typically open to allow PORV operation, the block valve may be OPERABLE when closed to isolate the flow path of an inoperable PORV because of excessive seat leakage. Isolation of an OPERABLE PORV does not render that PORV or block valve inoperable, provided the automatic pressure relief function remains available with timely operator actions (Ref. 1) to open the associated block valve, if closed, and assure the PORV's handswitch is in the automatic operation position. Satisfying the LCO helps minimize challenges to fission product barriers and precludes water relief through the pressurizer safety valves.

An OPERABLE PORV must not be experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criterion, exists when conditions dictate closure of the block valve to limit leakage.

(continued)

BASES (Continued)

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are required to be OPERABLE in MODES 1, 2, and 3 for automatic pressure relief to fulfill the required function of minimizing challenges to the pressurizer safety valves during an inadvertent ECCS actuation event. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for COMS in MODES 4 (with any RCS cold leg temperature ≤ 275°F), 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

A Note has been added to clarify that all pressurizer PORVs and block valves are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

A.1

The PORVs may be inoperable because of excessive seat leakage yet capable of automatic pressure relief and capable of being manually cycled. In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valves must be closed, but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. Credit for automatic PORV operation is taken in the Reference 1 safety analysis. However, the PORVs are considered OPERABLE in either the manual or automatic mode, as long as the automatic actuation circuitry is OPERABLE and the PORV can be made available for automatic pressure relief by timely operator actions (Ref. 1). Although a PORV may be designated inoperable, it may be available for automatic pressure relief and capable of being manually opened and closed and, therefore, able to perform its required safety functions.

(continued)

ACTIONS

A.1 (continued)

PORV inoperability solely due to excessive seat leakage does not prevent automatic and manual use and does not create a possibility for a small break LOCA. Closure of the block valve(s) establishes reactor coolant pressure boundary (RCPB) integrity for a PORV(s) with excessive seat leakage. RCPB integrity takes priority over the capability of the PORV(s) to mitigate an overpressure event. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE and automatic actuation status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

If one PORV is inoperable for reasons other than excessive seat leakage (i.e., not capable of automatic pressure relief or not capable of being manually cycled), it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times for Required Actions B.1 and B.2 of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable PORV cannot be restored to OPERABLE status, it must be isolated within the specified time of 1 hour. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. If the PORV cannot be restored within this additional time, the plant must be brought to MODE 4, as required by Condition D.

(continued)

ACTIONS (continued)

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of mitigating an event if the inoperable block valve is not fully open. If the block valve is restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation. If it cannot be restored within this additional time, the plant must be brought to MODE 4, as required by Condition D.

The Required Actions are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of automatic pressure relief or not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunction(s)).

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 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

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

systems. In MODES 1, 2, 3, and 4 (with any RCS cold leg temperature ≤ 275°F), 5, and 6 (with the reactor vessel head on), automatic PORV OPERABILITY is required. See LCO 3.4.12 for requirements in MODES 4, 5, and 6.

E.1, E.2, E.3, and E.4

If more than one PORV is inoperable for reasons other than excessive seat leakage, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If one PORV is restored and one PORV remains inoperable, then the plant will remain in Condition B with the time clock started at the time the remaining PORV was discovered to be inoperable (separate Condition entry for the PORVs). If no PORVs are restored within the Completion Time, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 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. In MODES 1, 2, 3, and 4 (with any RCS cold leg temperature ≤ 275°F), 5, and 6 (with the reactor vessel head on), automatic PORV OPERABILITY is required. See LCO 3.4.12 for requirements in MODES 4, 5, and 6.

F.1

If more than one block valve is inoperable, it is necessary to restore at least one block valve within 2 hours. The Completion Time is reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

If one block valve is restored within 2 hours and one block valve remains inoperable, then the plant will remain in Condition C with the time clock started at the time the remaining block valve was discovered to be inoperable (separate Condition entry for the block valves).

(continued)

ACTIONS

F.1 (continued)

The Required Action is modified by a Note stating that Required Action F.1 does not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of automatic pressure relief or not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunction(s)).

G.1 and G.2

If the Required Actions of Condition F are not met, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 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. In MODES 1, 2, 3, and 4 (with any RCS cold leg temperature ≤ 275°F), 5, and 6 (with the reactor vessel head on), automatic PORV OPERABILITY is required. See LCO 3.4.12 for requirements in MODES 4, 5, and 6.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The Note modifies this SR by stating that it is not required to be performed with the block valve closed, in accordance with the Required Actions of this LCO. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. Operating experience has shown that these valves usually pass the Surveillance when performed at the required INSERVICE TESTING PROGRAM frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 15.5.1.
2. Regulatory Guide 1.32, February 1977.
3. FSAR, Section 15.2.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Cold Overpressure Mitigation System (COMS)

BASES

BACKGROUND

The COMS controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the COMS MODES, as approved by NRC for Callaway in Ref. 12.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires both safety injection pumps and one ECCS centrifugal charging pump to be incapable of injection into the RCS and isolating the accumulators. The normal charging pump (NCP) has been analyzed as capable of injecting during an overpressure transient and the analysis assumes the flow from one ECCS centrifugal charging pump and the NCP. The term “centrifugal charging pump” or “CCP” refers to the safety-related ECCS pumps only (PBG05A and PBG05B). The normal charging pump or NCP (PBG04) does not serve an ECCS function (the NCP is tripped by a safety injection signal). The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief

(continued)

BACKGROUND (continued)

valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the COMS MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one ECCS centrifugal charging pump for makeup in the event of loss of inventory, either the NCP or other ECCS pumps can be made available through manual actions.

The COMS for pressure relief consists of two PORVs with reduced lift settings, or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to prevent overpressurization for the required coolant input capability.

PORV Requirements

As designed for the COMS, each PORV is signaled to open if the RCS pressure approaches a limit determined by the COMS actuation logic. The COMS actuation logic monitors both RCS temperature and RCS pressure and determines when a condition not acceptable with respect to the PTLR limits is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The PTLR presents the setpoints for COMS. The setpoints are normally staggered so only one valve typically opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset

(continued)

BACKGROUND (continued)

pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RHR Suction Relief Valve Requirements

During COMS MODES, the RHR System is operated for decay heat removal and low pressure letdown control. Therefore, the RHR suction isolation valves are open in the piping from the RCS hot legs to the inlets of the RHR pumps. While these valves are open, the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.

The RHR suction isolation valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. The RHR suction relief valves are spring loaded, bellows type water relief valves with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting COMS mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with all RCS cold leg temperatures exceeding 275°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. In MODE 4 (with any RCS cold leg temperature ≤ 275°F) and below, overpressure prevention falls to two OPERABLE RCS relief valves or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor

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APPLICABLE SAFETY ANALYSES (continued)

vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the COMS must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the COMS requirements. Any change to the RCS must be evaluated against the analyses in References 9 and 12 to determine the impact of the change on the COMS acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

<u>Mass Input Type Transients</u>

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

<u>Heat Input Type Transients</u>

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required, with exceptions described below, during the COMS MODES to ensure that mass and heat input transients do not occur, which either of the COMS overpressure protection means cannot handle:

a. Rendering both safety injection pumps and one ECCS centrifugal charging pump incapable of injection (there are no limitations on the use of the NCP during the COMS MODES);

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Precluding start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6,

(continued)

APPLICABLE SAFETY ANALYSES (continued)

"RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection.

The analyses in References 9 and 12 demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when one ECCS centrifugal charging pump (in addition to the NCP) is actuated. Thus, the LCO allows one ECCS centrifugal charging pump and the NCP to be capable of injection during the COMS MODES. Since neither one RCS relief valve nor the RCS vent can handle the pressure transient caused by accumulator injection when RCS temperature is low, the LCO also requires accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

Fracture mechanics analyses established the temperature of COMS Applicability at 275°F, which is conservative with respect to the ASME Code Case N-514 limit of 200°F (Ref. 12).

<u>PORV Performance</u>

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limits shown in the PTLR. The setpoints are derived by analyses that model the performance of the COMS, assuming the mass injection transient of one ECCS centrifugal charging pump and the NCP injecting into the RCS and the heat injection transient of starting an RCP with the RCS 50°F colder than the secondary coolant. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensure
the Reference 1 P/T limits will be met.

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the COMS analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses show that one RHR suction relief valve with a setpoint at or between 436.5 psig and 463.5 psig will pass flow greater than that required for the limiting COMS transient while maintaining RCS pressure less than the P/T limit curve.

The RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement. The RHR suction relief valves must be analyzed to accommodate the design basis transients for COMS.

The RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 2.0 square inches is capable of mitigating the limiting COMS transient. The capacity of a vent this size is greater than the flow of the limiting transient for the COMS configuration, one ECCS centrifugal charging pump and the NCP injecting into the RCS, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The COMS satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

This LCO requires that the COMS is OPERABLE. The COMS is OPERABLE when the maximum specified coolant input and minimum pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

(continued)

LCO (continued)

To limit the coolant input capability, the LCO requires that a maximum of zero safety injection pumps, one ECCS centrifugal charging pump, and the NCP be capable of injecting into the RCS and all accumulator discharge isolation valves be closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The LCO is modified by three Notes. Note 1 allows two ECCS centrifugal charging pumps to be made capable of injecting for ≤ 1 hour for pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and surveillance requirements associated with the swap. The intent is to minimize the actual time that both ECCS centrifugal charging pumps, in addition to the NCP, are physically capable of injection.

Note 2 states that one or more safety injection pumps may be made capable of injecting in MODES 5 and 6 when the RCS water level is below the top of the reactor vessel flange for the purpose of protecting the decay heat removal function.

Note 3 states that the accumulator may be unisolated when the accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature, as allowed by the P/T limit curves provided in the PTLR. The accumulator discharge isolation valve Surveillance is not required under these pressure and temperature conditions.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs; or

   A PORV is OPERABLE for COMS when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

b. Two OPERABLE RHR suction relief valves; or

   An RHR suction relief valve is OPERABLE for COMS when its RHR suction isolation valves are open, its setpoint is at or between 436.5 psig and 463.5 psig, and testing has proven its ability to open at this setpoint.

(continued)

LCO (continued)

c. One OPERABLE PORV and one OPERABLE RHR suction relief valve; or

d. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of ≥ 2.0 square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting COMS transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ 275°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 275°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES.
LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 275°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable COMS. There is an increased risk associated with entering MODE 4 from MODE 5 with COMS inoperable and 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.

A.1 and B.1

With one or more safety injection pumps or two ECCS centrifugal charging pumps capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability (i.e., no safety injection pumps and a maximum of one ECCS CCP and the

(continued)

ACTIONS

A.1 and B.1 (continued)

NCP capable of injecting) to the RCS reflects the urgency of removing the RCS from this condition.

C.1, D.1, and D.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 275°F, an accumulator pressure of 648 psig cannot exceed the COMS limits if the accumulators are fully injected. Due to this Specification's Note restricting the application of LCO 3.0.4.b, Required Action D.1 can only be used if the plant is in MODE 4 when Conditions C and D are entered. Depressurizing the accumulators below the COMS limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring COMS is not likely in the allowed times.

E.1

In MODE 4 when any RCS cold leg temperature is ≤ 275°F, with one required RCS relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS relief valves in any combination of the PORVS and the RHR suction relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the RCS relief valves is required to mitigate an overpressure transient transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

(continued)

BASES

ACTIONS (continued)

F.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.

G.1

The RCS must be depressurized and a vent must be established within 12 hours when:

a. Both required RCS relief valves are inoperable; or

b. A Required Action and associated Completion Time of Condition A, B, D, E, or F is not met; or

c. The COMS is inoperable for any reason other than Condition A, B, C, D, E, or F.

The vent must be sized ≥ 2.0 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of zero safety injection pumps, one ECCS centrifugal charging pump, and the NCP are verified to be capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed with power removed from the valve

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (continued)

operators (Refs. 10 and 11). Verification that each accumulator is isolated is only required when accumulator isolation is required as stated in Note 3 to the LCO.

The safety injection pumps and one ECCS centrifugal charging pump are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of cold overpressure protection control may be employed using at least two independent means to render a pump incapable of injecting into the RCS such that a single failure or single action will not result in an injection into the RCS. This may be accomplished by placing the pump control switch in pull to lock and closing at least one valve in the discharge flow path, or by closing at least one valve in the discharge flow path and removing power from the valve operator, or by closing at least one manual valve in the discharge flow path under administrative controls.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.4

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the INSERVICE TESTING PROGRAM. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction isolation valves are verified to be opened. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The ASME Code (Ref. 8), test per INSERVICE TESTING PROGRAM verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.5

The RCS vent of ≥ 2.0 square inches is proven OPERABLE by verifying its open condition either:

a. For a vent pathway that is not locked, sealed, or otherwise secured in the open position, or

b. For a valve that is locked, sealed, or otherwise secured in the open position. A removed pressurizer safety valve or open manway also fits this category.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.6

Any passive vent path arrangement must only be open when required to be OPERABLE. This Surveillance is required if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12d.

The PORV block valve must be verified open to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is only required to be performed if the PORV is being used to meet this LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive seat leakage or does not close (sticks open) after relieving an overpressure situation.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.7

Not used.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.8

Performance of a COT is required on each required PORV to verify the PORV is capable of performing its COMS function and, as necessary, adjust its lift setpoint. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified periodically by other Technical Specifications and non-Technical Specifications tests. The COT will verify the BBTY0413M and BBTY0413P function generator card breakpoints are within the limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

A Note has been added indicating that this SR is not required to be performed until 12 hours after decreasing any RCS cold leg temperature to ≤ 275°F. The Note allows, but does not require, entering the COMS LCO Applicability prior to performing the SR. The 12 hour allowance considers the unlikelihood of a low temperature overpressure event during this time.

SR 3.4.12.9

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required periodically to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. FSAR, Chapter 15.

(continued)

REFERENCES (continued)

5. 10 CFR 50, Section 50.46.

6. 10 CFR 50, Appendix K.

7. Generic Letter 90-06.

8. ASME Code for Operation and Maintenance of Nuclear Power Plants.

9. FSAR Section 5.2.2.10.

10. FSAR Section 6.3.2.

11. FSAR Section 7.6.4.

12. Amendment No. 124 to Facility Operating License NPF-30 dated April 2, 1998.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can allow varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of RCS Operational LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS Operational LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

(continued)

BASES (Continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address RCS Operational LEAKAGE. However, the other forms of RCS Operational LEAKAGE are related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analyses for events resulting in steam discharge to the atmosphere assume that primary to secondary LEAKAGE through all steam generators (SGs) is one gallon per minute. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analyses.

Primary to secondary LEAKAGE is a factor in the activity releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involving secondary steam release to the atmosphere include the steam generator tube rupture (SGTR). The primary to secondary leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes that some of the contaminated secondary fluid is released via a postulated stuck-open atmospheric steam dump (ASD) valve or a partially stuck-open main steam safety valve. The 1 gpm primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential for SGTR given the magnitude of the postulated break flow rate.

The safety analysis for the SLB accident assumes the entire 1 gpm primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accidents involving secondary steam release to the atmosphere are within the limits defined in 10 CFR 50.67 (Ref. 5) and Regulatory Guide 1.183 (Ref. 9).

The safety analysis for RCS main loop piping for GDC-4 (Ref. 1) assumes 1 gpm unidentified leakage and monitoring per RG 1.45 (Ref. 2) are maintained (Ref. 4).

The RCS operational LEAKAGE satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued

(continued)

BASES

LCO

a. Pressure Boundary LEAKAGE (continued)

degradation of the RCPB. LEAKAGE past seals, gaskets, and instrumentation lines is not pressure boundary LEAKAGE. Instrumentation lines are 3/8 inch tubing for instrument connections to ASME Class 1 fluid piping downstream of the root valves and 1/8 inch core exit thermocouple sheaths. These instrument lines are not part of the reactor coolant pressure boundary (RCPB) nor do they provide a pressure retaining barrier. Normal charging can accommodate a 3/8 inch break and maintain normal pressurizer level such that the ECCS is not actuated.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the leak monitoring equipment in LCO 3.4.15, "RCS Leakage Detection Instrumentation," can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). RCP number 2 seal leak off is included in the measured identified LEAKAGE since it is directed to the RCDT along with other identified leakage sources. Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on Reference 6 and the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 7). 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 SG tube degradation mechanisms that result in tube leakage. The

(continued)

BASES

LCO

d. Primary to Secondary LEAKAGE Through Any One SG (continued)

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.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCS Operational LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, RCS Operational LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion 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 RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if primary to secondary LEAKAGE is not within limit, or if unidentified LEAKAGE 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, gaskets, and instrumentation lines is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS Operational LEAKAGE to be within the LCO limits ensures the integrity of the RCPB 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, gaskets, and instrumentation lines is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). SR 3.4.13.1 is modified by two Notes. Note 1 states that this SR 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 preferred to perform a proper inventory balance since calculations during non-steady state conditions must account for the changing parameters.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals, gaskets, and instrumentation lines is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

Note 2 states that SR 3.4.13.1 is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 8. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

SR 3.4.13.2 is modified by a Note which 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 RCP 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 the methodology of Reference 8. Leakage verification is provided by chemistry procedures that provide alternate means of calculating and confirming primary to secondary leakage is less than or equal to 150 gallons per day through any one SG.

(continued)

BASES (Continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.6.3.
4. NUREG-1061, Volume 3, November 1984.
5. 10 CFR 50.67.
6. Amendment No. 116 dated October 1, 1996.
7. NEI 97-06, "Steam Generator Program Guidelines."
8. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."
9. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS 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 3.4.13.1). 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 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.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System;

(continued)

BASES

BACKGROUND (continued)

b. Safety Injection System; and

c. Chemical and Volume Control System.

The PIVs are listed below :

VALVE NUMBER	VALVE SIZE (in.)	FUNCTION	MAXIMUM ALLOWABLE LEAKAGE (gpm)
BB8948A	10	RCS Loop 1 Cold Leg SI Accu Chck	5.0
BB8948B	10	RCS Loop 2 Cold Leg SI Accu Chck	5.0
BB8948C	10	RCS Loop 3 Cold Leg SI Accu Chck	5.0
BB8948D	10	RCS Loop 4 Cold Leg SI Accu Chck	5.0
BB8949B	6	RCS Loop 2 Hot Leg SI/RHR Pump Chck	3.0
BB8949C	6	RCS Loop 3 Hot Leg SI/RHR Pump Chck	3.0
BB8949D	6	RCS Loop 4 Hot Leg SI/RHR Pump Chck	3.0
BB8949E	2	RCS Loop 1 Hot Leg SI/RHR Pump Chck	1.0
BBV0001	1.5	RCS Loop 1 Cold Leg SI/Boron Injection Header Chck	0.75
BBV0022	1.5	RCS Loop 2 Cold Leg SI/Boron Injection Header Chck	0.75
BBV0040	1.5	RCS Loop 3 Cold Leg SI/Boron Injection Header Chck	0.75
BBV0059	1.5	RCS Loop 4 Cold Leg SI/Boron Injection Header Chck	0.75
BBPV8702A	12	RCS Loop 1 Hot Leg to RHR Pumps ISO	5.0
BBPV8702B	12	RCS Loop 4 Hot Leg to RHR Pumps ISO	5.0
EJ8841A	6	RHR TRNS SIS Hot Leg Loop 2 Recirc	3.0
EJ8841B	6	RHR TRNS SIS Hot Leg Loop 3 Recirc	3.0
EJHV8701A	12	RHR Pump A Suction ISO	5.0
EJHV8701B	12	RHR Pump B Suction ISO	5.0
EMV0001	2	SI Pump A Disch to Hot Leg Loop 2 Chck	1.0
EMV0002	2	SI Pump A Disch to Hot Leg Loop 3 Chck	1.0
EMV0003	2	SI Pump B Disch to Hot Leg Loop 1 Chck	1.0
EMV0004	2	SI Pump B Disch to Hot Leg Loop 4 Chck	1.0
EM8815	3	Boron Injection Header CVCS Out Check	1.5
EPV0010	2	SI Pumps to RCS Cold Leg Loop 1 Chck	1.0
EPV0020	2	SI Pumps to RCS Cold Leg Loop 2 Chck	1.0
EPV0030	2	SI Pumps to RCS Cold Leg Loop 3 Chck	1.0
EPV0040	2	SI Pumps to RCS Cold Leg Loop 4 Chck	1.0
EP8818A	6	RHR Pumps to RCS Cold Leg Loop 1 Chck	3.0
EP8818B	6	RHR Pumps to RCS Cold Leg Loop 2 Chck	3.0
EP8818C	6	RHR Pumps to RCS Cold Leg Loop 3 Chck	3.0

(continued)

BASES

BACKGROUND (continued)

VALVE NUMBER	VALVE SIZE (in.)	FUNCTION	MAXIMUM ALLOWABLE LEAKAGE (gpm)
EP8818D	6	RHR Pumps to RCS Cold Leg Loop 4 Chck	3.0
EP8956A	10	SI Accu TK A Out Upstream Chck	5.0
EP8956B	10	SI Accu TK B Out Upstream Chck	5.0
EP8956C	10	SI Accu TK C Out Upstream Chck	5.0
EP8956D	10	SI Accu TK D Out Upstream Chck	5.0

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm.

(continued)

BASES

LCO (continued)	Reference 6 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.
APPLICABILITY	In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation. In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.
ACTIONS	The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability. <u>A.1 and A.2</u> The flow path must be isolated by two valves. Required Action A.1 is modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB. Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves. Required Action A.2 specifies that the double isolation barrier of two valves be restored by restoring the RCS PIV to within limits. The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This time frame considers the time required to complete the Action and the low probability of a second valve failing during this time period.

(continued)

ACTIONS (continued)

B.1 and B.2

If leakage cannot be reduced or the system isolated, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. 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.

C.1

The inoperability of the RHR suction isolation valve interlock could allow inadvertent opening of the valves at RCS pressures in excess of the RHR system design pressure. If the RHR suction isolation valve interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one deactivated remote manual valve within 4 hours. This Action accomplishes the purpose of the interlock.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV used to satisfy Required Action A.1 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 9 months, but may be extended if the plant does not go into MODE 5 for at least 7 days. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

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SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

Test pressures less than 2235 psig but greater than 150 psig are allowed for valves where higher pressures would tend to diminish leakage channel opening. Observed leakage shall be adjusted for actual pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

In addition, testing must be performed once after the check valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the check valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a check valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

SR 3.4.14.2

The RHR suction isolation valve interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < 425 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. This Surveillance does not have to be met when the RHR suction relief valves

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.2 (continued)

are used to satisfy LCO 3.4.12. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. ASME Code for Operation and Maintenance of Nuclear Power Plants.
7. 10 CFR 50.55a(g).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump level and flow monitoring system, used to collect unidentified LEAKAGE, and containment cooler condensate monitoring system are instrumented to alarm for increases of 0.5 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivity of 10^{-9} μCi/cc radioactivity for particulate monitoring is practical for this leakage detection system. This radioactivity detection system is included for monitoring particulate activities because of its sensitivity and rapid response to RCS LEAKAGE.

The measurement of containment atmosphere gaseous radioactivity is less sensitive than the measurement of particulate radioactivity for the purpose of detecting RCS leakage. Evaluations have shown that the pre-existing containment radioactive gaseous background levels for which reliable detection is possible is dependent upon the reactor power level, percent failed fuel in the reactor, and air volume exchange brought about by the containment purge system. With primary coolant concentrations less than equilibrium levels, such as during reactor startup and operation

(continued)

BASES

BACKGROUND (continued)

with no fuel defects, the increase in detector count rate due to leakage will be partially masked by the statistical variation of the minimum detector background count rate, rendering reliable detection of a 1 gpm leak uncertain. The containment gaseous radioactivity monitor is considered most useful for detecting an RCS-to-containment atmosphere leak if elevated reactor coolant gaseous activity is present. The containment gaseous radioactivity monitors are not required by this LCO. (Reference 7)

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew point is well within the sensitivity range of available instruments.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump and condensate flow from air coolers. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements is affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The asymmetric loads produced by postulated breaks are the result of assumed pressure imbalance, both internal and external to the RCS. The internal asymmetric loads result from a rapid decompression that causes large transient pressure differentials across the core barrel and fuel assemblies. The external asymmetric loads result from the rapid depressurization of the annulus regions, such as the annulus between the reactor vessel and the shield wall, and cause large transient pressure differentials to act on the vessel. These differential pressure loads could damage RCS supports, core cooling equipment or core internals. This concern was first identified as Multiplant Action (MPA) D-10 and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

subsequently as Unresolved Safety Issue (USI) 2, "Asymmetric LOCA Loads" (Ref. 4).

The resolution of USI-2 for Westinghouse PWRs was the use of fracture mechanics technology for RCS piping > 10 inches diameter (Ref. 5). This technology became known as leak-before-break (LBB). Included within the LBB methodology was the requirement to have leak detection systems capable of detecting a 1.0 gpm leak within four hours. This leakage rate is designed to ensure that adequate margins exist to detect leaks in a timely manner during normal operation conditions. Actual leakage detection capabilities are discussed in Reference 3.

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The individual system response times and sensitivities are described in the FSAR (Ref. 3). Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10CFR50.36(c)(2)(ii).

LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

This LCO is satisfied when diverse monitoring methods are available. Thus, the containment sump level and flow monitoring system, one containment atmosphere particulate radioactivity monitor, and the containment cooler condensate monitoring system provide an acceptable minimum. For the containment atmosphere particulate radioactive

(continued)

BASES

LCO (continued)

monitor, particulate channels of either GTRE0031 or GTRE0032 satisfy the LCO requirement.

The sump level and flow monitoring system, the containment air particulate monitoring system, and the containment cooler condensate measuring system are capable of detecting a one gpm leak in one hour at the sensitivity recommended in Regulatory Guide 1.45.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is required to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

A primary system leak would result in reactor coolant flowing into the containment normal sumps or into the instrument tunnel sump. Indication of increasing sump level is transmitted to the control room by means of individual sump level transmitters. This information is used to provide the measurement of low leakage by monitoring level increase versus time.

With the required containment sump level and flow monitoring system inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere particulate radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required sump level and flow monitoring system to OPERABLE status within a Completion Time of 30 days is required to

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

regain the function after the system's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

B.1.1, B.1.2, B.2.1, and B.2.2

With the containment atmosphere particulate radioactivity monitoring instrumentation channel inoperable, alternative action is required. Either samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Samples of the containment atmosphere are obtained and analyzed for particulate radioactivity.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere particulate radioactivity monitor. Alternatively, continued operation is allowed if the containment air cooler condensate monitoring system is OPERABLE, provided grab samples are taken or water inventory balances are performed every 24 hours.

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

C.1 and C.2

With the required containment cooler condensate monitoring system inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or a water inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment cooler condensate monitoring system to OPERABLE status.

(continued)

ACTIONS

C.1 and C.2 (continued)

The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (near operating rated operating pressure with stable RCS pressure, temperature, power level, pressurizer and makeup tank level, makeup and letdown, and RCP seal injection and return flows.) The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

D.1, and D.2

With the required containment atmosphere particulate radioactivity monitor and the required containment cooler condensate monitoring system inoperable, the only means of detecting leakage is the containment sump level and flow monitoring system. This Condition does not provide all the required diverse means of leakage detection. The Required Action is to restore either of the inoperable required monitoring methods to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

E.1 and E.2

If a Required Action of Condition A, B, C, or D cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 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.

F.1

With all required monitoring methods inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

(continued)

BASES (Continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere particulate radioactivity monitors (GTRE0031 or GTRE0032). The check gives reasonable confidence that the channels are operating properly. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The RM-23 unit display must be used to perform the CHANNEL CHECK.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere particulate radioactivity monitors (GTRE0031 or GTRE0032). The test ensures that either monitor can perform its function in the desired manner. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified periodically by other Technical Specifications and non-Technical Specifications tests. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. During performance of the COT, verification of the RM-23 unit display and alarm functions is required.

SR 3.4.15.3, SR 3.4.15.4, and SR 3.4.15.5

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. During performance of the CHANNEL CALIBRATION for the required containment atmosphere particulate radioactivity monitors (GTRE0031 or GTRE0032), verification of the RM-23 unit display and alarm functions is required.

(continued)

BASES (Continued)

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.

2. Regulatory Guide 1.45.

3. FSAR Section 5.2.5.

4. NUREG-609, "Asymmetric Blowdown Loads on PWR Primary Systems," 1981.

5. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops. "

6. FSAR Table 11.5-3

7. NRC Letter, “Callaway Plant, Unit 1 - License Amendment Request to Change the Reactor Coolant System Leakage Detection Instrumentation Methodology (TAC No. MC8220), May 16, 2006.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose that an individual at the exclusion area boundary can receive for any 2-hour time period following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1) and Regulatory Guide 1.183 (Ref. 2). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in Ref. 1 and Ref. 2.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following an SLB and SGTR accident. The safety analyses (Refs. 3 and 4) assume the initial iodine specific activity of the reactor coolant is greater than the LCO limit (see the discussion of Case 1 below), and a pre-accident reactor coolant steam generator (SG) tube leakage rate of 1 gpm exists. The safety analyses assume the initial iodine specific activity of the secondary coolant is at the limit of 0.1 μCi/gm DOSE EQUIVALENT I -131 from LCO 3.7.18, “Secondary Specific Activity.”

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the plant that could affect RCS specific activity, as they relate to the acceptance limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The safety analyses consider two cases of reactor coolant iodine specific activity. In Case 1, the initial reactor coolant iodine specific activity corresponds to an isotope mixture that bounds the SR 3.4.16.2 limit for both tight and open fuel defects. The isotopic mix is based on the initial RCS concentrations from FSAR Table 15A-5. This table provides conservative values for the iodine isotopic spectrum that bound the RCS concentrations which could be expected with either tight or open fuel defects. Since the assumed iodine spectrum represents bounding values for different types of fuel defects, the initial radioiodine inventory exceeds the SR 3.4.16.2 limit of 1.0 μCi/gm.

Case 1 also assumes an accident-initiated iodine spike that increases the rate of iodine release from the fuel rods containing cladding defects to the primary coolant immediately after an SLB or SGTR. The iodine spiking factor is assumed to be 500 for the Case 1 radiological consequence evaluations for SLB and 335 for the Case 1 radiological consequence evaluation for both SGTR radiological consequence analyses.

Case 2 radiological consequence evaluations for SLB and SGTR assume the initial reactor coolant iodine specific activity is a factor of 60 higher than Case 1 due to a pre-accident iodine spike caused by a transient prior to the accident.

In both Case 1 and Case 2 radiological consequence evaluations, the noble gas specific activity in the reactor coolant is assumed to be greater than the 225 μCi/gm DOSE EQUIVALENT XE-133 limit in SR 3.4.16.2. The dose analysis assumptions are discussed further in Tables 15.1-3 and 15.6-4 of Reference 4.

The SGTR analysis also assumes a loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal in the analysis of an SGTR with a failed ASD on the faulted steam generator. In the analysis of an SGTR with a failed AFW flow control valve on the faulted steam generator, reactor trip and safety injection are assumed to occur at the time of the tube rupture to maximize the potential for overfilling the ruptured steam generator.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG atmospheric steam dump valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steamline pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible if the activity levels do not exceed 60 μCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

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 225 μCi/gm DOSE EQUIVALENT XE - 133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) 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 an SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

APPLICABILITY

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 an SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

(continued)

BASES (Continued)

ACTIONS

A.1 and A.2

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 $\leq 60\ \mu$Ci/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue 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 an SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions A.1 ad A.2 while the DOSE EQUIVALENT I-131 LCO limit 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.

B.1

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 an SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit 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.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is >60.0 μCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The 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.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant. 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 Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 in Specification 1.1, “Definitions,” is not detected, it should be assumed to be present at the minimum detectable activity.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

SR 3.4.16.2

This Surveillance 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 Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The Frequency, between 2 and 6 hours after a power change ≥ 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.16.2 (continued) The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.
REFERENCES	1. 10 CFR 50.67. 2. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000. 3. FSAR, Section 15.1.5. 4. FSAR, Section 15.6.3.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity; accident induced leakage; and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

(continued)

BASES (Continued)

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes, and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a primary to secondary LEAKAGE rate of 1 gpm to the unaffected steam generators, exceeding the operational LEAKAGE rate limits in LCO 3.4.13, “RCS Operational LEAKAGE,” plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes that some of the contaminated secondary fluid is released to the atmosphere via a postulated stuck-open atmospheric steam dump (ASD) valve or via a partially stuck-open main steam safety valve (see Ref. 2).

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, “RCS Specific Activity,” limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 3), 10 CFR 50.67 (Ref. 4), and Regulatory Guide 1.183 (Ref. 8).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

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 a 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.

(continued)

BASES

LCO (continued)

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "Steam Generator 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 follows: 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 have a significant effect on burst or collapse. In that context, the term significant is defined as follows: 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 steam generator 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 (Ref. 5) and Draft Regulatory Guide 1.121 (Ref. 6).

(continued)

BASES

LCO (continued)

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm total for all four steam generators. 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 3.4.13, “RCS 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.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies 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 as required by SR 3.4.17.2. An evaluation of SG tube integrity of the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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 degradation 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. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion 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, Required Action A.2 allows plant operation 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 MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion 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.

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BASES (Continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), 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, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 7). 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 5.5.9 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 5.5.9 until subsequent inspections support extending the inspection interval.

SR 3.4.17.2

During a 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 5.5.9 are intended to

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.2 (continued)

ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw 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). Reference 1 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 MODE 4 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.

REFERENCES

1. NEI 97-06, “Steam Generator Program Guidelines.”
2. FSAR Section 15.6.3.
3. 10 CFR 50 Appendix A, GDC 19.
4. 10 CFR 50.67.
5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
6. Draft Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.
7. EPRI TR-107569, “Pressurized Water Reactor Steam Generator Examination Guidelines.”
8. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.