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 (Reference 1) of normal operating conditions and anticipated operational occurrences, assume initial conditions within the normal steady-state envelope. The limits placed on departure from nucleate boiling (DNB) related parameters ensure that these parameters will not be less conservative than were assumed in the analyses, and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The Limiting Condition for Operation (LCO) limit for minimum RCS pressure as measured at the pressurizer is consistent with operation within the nominal operating envelope and is bounded by the initial pressure in the analyses.

The LCO limit for maximum RCS cold leg temperature is consistent with operation at the indicated power level and is bounded by the initial temperature in the analyses.

The LCO limit for minimum RCS flow rate is bounded by the initial flow rate in the analyses. The RCS flow rate is not expected to vary during plant operation with all pumps running.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the safety analyses (Reference 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed include loss of coolant flow events and dropped or stuck control element assembly events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, LCO 3.2.4, and LCO 3.2.5.

BASES

The RCS DNB limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

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

The LCO numerical values for pressure and temperature (P/T) specified in the COLR are given for the measurement location and have been adjusted for instrument error. Reactor Coolant System flow rate specified in the COLR is given as an analytical value.

APPLICABILITY

In MODE 1, the limits on RCS pressurizer pressure, RCS cold leg temperature, and RCS flow rate must be maintained during steady-state operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODEs, the power level is low enough so that DNBR is not a concern.

A Note has been added to indicate the limit on pressurizer pressure may be exceeded during short-term operational transients such as a THERMAL POWER ramp increase of > 5% RATED THERMAL POWER (RTP) per minute or a THERMAL POWER step increase of > 10% RTP. These conditions represent short-term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in Safety Limit (SL) 2.1.1. Those limits are less restrictive than the limits of this LCO, but violation of SLs merits a stricter, more severe Required Action. Should

BASES

a violation of this LCO occur, the operator should check whether or not an SL may have been exceeded.

ACTIONS

A.1

Pressurizer pressure and RCS cold leg temperature are controllable and measurable parameters. Reactor Coolant System flow rate is not a controllable parameter and is not expected to vary during steady-state operation. With any parameter not within its LCO limit, action must be taken to restore the parameter.

The two hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause of the off normal condition, and to restore the readings within limits. The Completion Time is based on plant operating experience that shows the parameter can be restored in this time period.

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 six hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

Six hours is a reasonable time that permits the plant power to be reduced at an orderly rate in conjunction with even control of steam generator (SG) heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of two hours to restore parameters that are not within limits, the 12 hour SR Frequency for pressurizer pressure is sufficient to ensure that the pressure can be restored to a normal operation, steady-state condition following load changes, and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of two hours to restore parameters that are not within limits, the 12 hour SR Frequency for cold leg temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady-state condition following load changes, and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions. Average Reactor Protective System (RPS) cold leg indication may be used for this SR as described in plant procedures. Use of the maximum RPS cold leg indication for this SR is acceptable and conservative at all times.

SR 3.4.1.3

The 12 hour SR Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour Frequency has been shown by operating experience to be sufficient to assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate is performed once every 24 months. This verifies that the actual RCS flow rate is within the bounds of the analyses.

The Frequency of 24 months reflects the importance of verifying flow after a refueling outage where the core has been altered, which may have caused an alteration of flow resistance.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR), Section 14.1.2, "Plant Characteristics Considered in Safety Analysis"

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for:

a. Operation within the existing instrumentation ranges and accuracies;

b. Operation within the bounds of the existing accident analyses; and

c. Operation with the reactor vessel above its minimum nil ductility reference temperature when the reactor is critical.

The reactor coolant moderator temperature coefficient used in core operating and accident analysis is defined for the normal operating temperature range, as specified in the operating procedures. The Reactor Protective System (RPS) receives inputs from the narrow range hot and cold leg temperature detectors, which have a range of 515°F to 665°F and 465°F to 615°F, respectively. The RCS temperature is controlled using inputs of the same range. Nominal T_{avg} for making the reactor critical is 532°F. Safety and operating analyses for lower temperature have not been made.

APPLICABLE SAFETY ANALYSES

The Safety Analyses initiated from Hot Zero Power that assume a minimum RCS temperature as an initial condition use 515°F, which is the Technical Specification minimum temperature for criticality (Reference 1).

The RCS minimum temperature for criticality satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The purpose of the LCO is to prevent criticality outside the normal operating regime and to prevent operation in an unanalyzed condition.

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODEs 1 and 2 only and in accordance with this specification. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2 when $K_{eff} \geq 1.0$.

BASES

ACTIONS

A.1

If T_{avg} is below 515°F, 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 reflects the ability to perform this action and to maintain the plant within the analyzed range.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

T_{avg} is initially required to be verified ≥ 515°F within 30 minutes prior to reaching reactor criticality, then T_{avg} is required to be verified ≥ 515°F every 30 minutes. The 30 minute time period is frequent enough to prevent inadvertent violation of the LCO. The second frequency is modified by a Note which states that the surveillance test is only required to be performed when RCS T_{avg} is less than 525°F. This provides a reasonable distance to the limit of 515°F. Adequate time will be available to trend RCS T_{avg} as it approaches 515°F, and take corrective action(s) prior to exceeding the limit.

REFERENCES

1. UFSAR, Chapter 14, "Safety Analysis"

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 P/T changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the P/T changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.3-1 and 3.4.3-2 contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (these P/T limits do not apply to the pressurizer).

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 P/T 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, which has different design characteristics and operating functions.

Reference 1, Appendix G .requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1, Appendix G requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the Reference 2, Section III, Appendix G.

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

BASES

with References 1 (Appendix H) and 3. The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 2, Section III, Appendix G.

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.

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 can alter the location of the tensile stress between the outer and inner walls.

The criticality limit includes the Reference 1, Appendix G requirement that the limit be no less than 40°F above the heatup curve or the cooldown curve and not less than the minimum permissible temperature for the ISLH testing. However, the criticality limit is not operationally limiting; a more restrictive limit exists in LCO 3.4.2.

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 (LOCA). In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. Reference 2, Section XI, Appendix E, 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. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits

BASES

themselves, since they preclude operation in an unanalyzed condition.

The RCS P/T limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

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.

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.

The P/T limits are corrected for instrument uncertainty, and for static and dynamic head between the limiting material location and the pressurizer. The limits assume not more than the following number of RCPs are running:

Heatup

RCS Temperature (Unit 1)	RCS Temperature (Unit 2)	Number of RCPs
70°F to 330°F	70°F to 308°F	2
> 330°F	> 308°F	4

BASES

Cooldown

RCS Temperature (Unit 1)	RCS Temperature (Unit 2)	Number of RCPs
> 350°F	> 350°F	4
350°F to 150°F	350°F to 150°F	2
< 150°F	< 150°F	0

Violating 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 follows:

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 specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with Reference 1, Appendix G. 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.

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, LCO 3.4.2, and SL 2.1, also provide operational restrictions for P/T and maximum 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.

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from Emergency Operating Procedures.

ACTIONS

A.1 and A.2

Operation outside the P/T limits 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 to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable by determining the effects of the out of limit condition on the fracture toughness properties of the RCS 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.

Reference 2, Section XI, Appendix E 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.

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:

a. The RCS remained in an unacceptable P/T region for an extended period of increased stress; or

b. 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 P/T. With reduced P/T conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in MODE 3 within 6 hours and in MODE 5 with RCS pressure < 300 psia 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.

C.1 and C.2

The actions of this LCO, anytime other than in MODEs 1, 2, 3, or 4, consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from Emergency Operating Procedures. Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The Completion Time of "immediately" 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 a short period of time in a controlled manner.

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

Reference 2, Section XI, Appendix E, may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The Completion Time of prior to entering MODE 4 forces the evaluation prior to entering a MODE where temperature and pressure can be significantly increased. The evaluation for a mild violation is possible within several days, but more severe violations may require special, event specific stress analyses or inspections.

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 limits is required every 30 minutes when RCS P/T conditions are undergoing planned changes. This Frequency is considered reasonable in view of the Control Room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

The SR 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 Surveillance Requirement (SR) is modified by a Note that requires this SR be performed only during RCS 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. 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities
2. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code
3. American Society for Testing Materials E 185-82, July 1982

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 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 RCS configuration for heat transport uses two RCS loops. Each RCS loop contains a SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to DNB during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with both RCPs in operation in each loop. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying two RCS loops provides the minimum necessary paths (two SGs) for heat removal.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the DBA 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.

Both transient and steady-state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming four RCPs are in operation. The majority of the plant

BASES

safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to RCP operation are loss of coolant flow and seized rotor (Reference 1).

RCS Loops - MODEs 1 and 2 satisfy 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG that is OPERABLE. Steam generator, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the RPS in MODEs 1 and 2. A reactor trip places the plant in MODE 3 if any SG level is $\geq$ 50 inches below normal water level as sensed by the RPS. The minimum water level to declare the SG OPERABLE is $<$ 50 inches below normal water level.

APPLICABILITY

In MODEs 1 and 2, the reactor is critical and thus 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, 5, and 6.

Operation in other MODEs is covered by: LCO 3.4.5, LCO 3.4.6, LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

BASES

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. It should be noted that the reactor will trip and place the plant in MODE 3 as soon as the RPS senses less than 370,000 gpm** RCS flow.

The Completion Time of six 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 every 12 hours of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which help to ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, Control Room indication and alarms will normally indicate loop status.

REFERENCES

1. UFSAR, Chapter 14, "Safety Analysis"

** The Reactor Coolant System Flow Rate limit shall be ≥ 340,000 gpm through Unit 2, Cycle 14.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

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

In MODE 3, RCPs are used to provide forced circulation heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP is sufficient to remove core decay heat. However, two RCS loops (i.e., RCS loop Nos. 11 and 12 for Unit 1 and RCS loop Nos. 21 and 22 for Unit 2) are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE.

Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured.

APPLICABLE SAFETY ANALYSES

Failure to provide 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 DBA or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

Reactor Coolant System Loops - MODE 3 satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The purpose of this LCO is to require two RCS loops to be available for heat removal, thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable (> -50 inches water level) of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation.

Note 1 permits a limited period of operation without RCPs. All RCPs may not be in operation for ≤ 1 hour per eight hour period and ≤ 2 hours per eight hour period for low flow testing. This means that natural circulation has been established. When in natural circulation, a reduction in boron concentration with water at a boron concentration less than required to assure that the SHUTDOWN MARGIN (SDM) of LCO 3.1.1 is maintained, is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODE 3, it is sometimes necessary to stop all RCPs (e.g., to perform surveillance or startup testing). The time period is acceptable because natural circulation is adequate for heat removal and the reactor coolant temperature can be maintained subcooled.

Note 2 requires that all of the following three conditions be satisfied before an RCP can be started when any RCS cold leg temperature is ≤ 365°F (Unit 1), ≤ 301°F (Unit 2):

a. the pressurizer water level is ≤ 170 inches;

b. the pressurizer pressure is ≤ 300 psia (Unit 1), ≤ 320 psia (Unit 2); and

c. the secondary water temperature of each SG is ≤ 30°F above the RCS temperature. The RCS temperature used for this ΔT evaluation is the average RCS temperature. It may be conservatively measured using the cold leg, SDC return to the RCS, or, more accurately, the average RCS temperature depending on conditions. Where the measurement is taken is controlled by plant procedures.

Ensuring the above conditions are satisfied will preclude a power-operated relief valve (PORV) from opening as a result of the pressure surge in the RCS, when an RCP is started.

An OPERABLE loop consists of at least one OPERABLE RCP and an SG that is OPERABLE. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow, if required.

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

Operation in other MODEs is covered by: LCO 3.4.4, LCO 3.4.6, LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within a 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.

B.1

If restoration is not possible within 72 hours, the unit must be placed in MODE 4 within 12 hours. In MODE 4, the plant may be placed on the Shutdown Cooling (SDC) System. The Completion Time of 12 hours is compatible with required operation 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 no RCS loop is in operation, except as provided in Note 1 in the LCO section, all operations involving introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 must be immediately suspended. Action to restore one RCS loop to OPERABLE status and operation shall be initiated immediately and continued until one RCS loop is restored to OPERABLE status and operation. Suspending the introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. When water is added without forced circulation, unmixed coolant could be introduced to the core, however water added with a boron concentration meeting the minimum SDM maintains an acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required number of RCS loops are in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, Control Room indication and alarms will normally indicate loop status.

SR 3.4.5.2

This SR requires verification every 12 hours that the secondary side water level in each SG is > -50 inches. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within the safety analyses assumptions.

SR 3.4.5.3

Verification that the required number of RCPs are OPERABLE ensures that the single failure criterion is met and that an additional RCS loop 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 Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

BASES

REFERENCES None

3.4 REACTOR COOLANT SYSTEM (RCS)

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 transfer of this heat to the SGs or SDC heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either RCPs or SDC loops can be used for coolant circulation. The intent of this LCO is to provide forced flow from at least one RCP or one SDC loop for decay heat removal and transport. The flow provided by one RCP or SDC loop is adequate for heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for heat removal. For Unit 1, the two paths can be any combination of RCS loop No. 11, RCS loop No. 12, SDC loop No. 11, or SDC loop No. 12. For Unit 2, the two paths can be any combination of RCS loop No. 21, RCS loop No. 22, SDC loop No. 21, or SDC loop No. 22.

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 RCS and SDC loops provide this circulation.

Reactor Coolant System loops - MODE 4 have been identified in 10 CFR 50.36(c)(2)(ii) as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least two loops, RCS or SDC, be OPERABLE in MODE 4, and 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 and SDC System 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 and SDC pumps to not be in operation $\leq$ 1 hour per eight hour period. The Note prohibits boron dilution with water at a boron concentration less than that required to assure the SDM of LCO 3.1.1 is maintained when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the RCS without the RCPs or SDC pumps depends on the core decay heat load and the length of time that the pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits [P/T limits or low temperature overpressure protection (LTOP) limits] must be observed and forced SDC flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both RCPs or SDC pumps are to be limited to situations where:

a. Pressure and temperature increases can be maintained well within the allowable pressure (P/T limits and LTOP) and 10°F subcooling limits; or

b. An alternate heat removal path through the SGs is in operation.

Note 2 requires that the following conditions be satisfied before an RCP may be started with any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2):

a. Pressurizer water level is ≤ 170 inches;

b. Pressurizer pressure is ≤ 300 psia (Unit 1), ≤ 320 psia (Unit 2); and

c. Secondary side water temperature in each SG is ≤ 30°F above the RCS temperature. The RCS temperature used for this ΔT evaluation is the average RCS temperature. It may be conservatively measured using the cold leg, SDC return to the RCS, or, more accurately, the average RCS temperature depending on conditions. Where the measurement is taken is controlled by plant procedures.

Satisfying the above conditions will preclude a PORV from opening due to a pressure surge in the RCS when the RCP is started.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE and has the minimum water level specified in SR 3.4.6.2.

Similarly, for the SDC System, an OPERABLE SDC loop is composed of the OPERABLE SDC pump(s) capable of providing forced flow to the SDC heat exchanger(s). Reactor coolant pumps and SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs, or the SDC System.

Operation in other MODEs is covered by: LCO 3.4.4, LCO 3.4.5, LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1

If only one required RCS loop is OPERABLE and in operation, and no SDC loops are OPERABLE, redundancy for heat removal is lost. Action must be initiated immediately to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for decay heat removal.

B.1

If one required SDC loop is OPERABLE and in operation and no RCS loops are OPERABLE, redundancy for heat removal is lost. The plant must be placed in MODE 5 within the next 24 hours. Placing the plant in MODE 5 is a conservative action with regard to decay heat removal. With only one SDC loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining SDC loop, it would be safer to initiate that loss from MODE 5 ($\leq$ 200°F) rather than MODE 4 (> 200°F to < 300°F). The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 from MODE 4, with only one SDC loop operating, in an orderly manner and without challenging plant systems.

C.1 and C.2

If no RCS or SDC loops are OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1, must be suspended and action to restore one RCS or SDC loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. When water is added without forced circulation, unmixed coolant could be introduced to the core, however water added with a boron concentration meeting the minimum SDM maintains an acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of decay heat removal. The action to restore must continue until one loop is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one required loop is in operation. This ensures forced flow is providing heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, Control Room indication and alarms will normally indicate loop status.

SR 3.4.6.2

This SR requires verification every 12 hours of secondary side water level in the required SG(s) $>$ -50 inches. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

BASES

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or SDC loop 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 required loop components that are not in operation. For an RCS loop, the required component is a pump. For an SDC loop, the required components are the pump and valves. The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES None

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 the transfer of this heat either to the SG secondary side coolant, or the component cooling water via the SDC heat exchangers. While the principal means for decay heat removal is via the SDC System, the SGs 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 side 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. Due to the non-condensable gasses that come out of solution and restrict flow through the SG tubes, the SGs can only be credited when the RCS is capable of being pressurized. 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 SDC loops are the principal means for decay heat removal. 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 SDC loop for decay heat removal and transport. The flow provided by one SDC 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 decay heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an SDC loop (i.e., SDC loop No. 11 or No. 12 for Unit 1, and SDC loop No. 21 or No. 22 for Unit 2) that must be OPERABLE and in operation. The second path can be another OPERABLE SDC loop (i.e., SDC loop No. 11 or No. 12 for Unit 1, and SDC loop No. 21 or No. 22 for Unit 2), or through the SGs, each having an adequate water level.

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 SDC loops provide this circulation.

Reactor Coolant System loops - MODE 5 (Loops Filled) have been identified in 10 CFR 50.36(c)(2)(ii) as important contributors to risk reduction.

LCO

The purpose of this LCO is to require at least one of the SDC loops be OPERABLE and in operation with an additional SDC loop OPERABLE, or secondary side water level of each SG shall be ≥ -50 inches. One SDC loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second SDC loop is normally maintained OPERABLE as a backup to the operating SDC loop, to provide redundant paths for decay heat removal. However, if the standby SDC loop is not OPERABLE, a sufficient alternate method to provide redundant paths for decay heat removal is two SGs with their secondary side water levels ≥ -50 inches. Should the operating SDC loop fail, the SGs could be used to remove the decay heat.

Note 1 permits all SDC pumps to not be in operation ≤ 1 hour per eight hour period. The circumstances for stopping both SDC loops are to be limited to situations where P/T increases can be maintained well within the allowable pressure (P/T and LTOP) and 10°F subcooling limits, or an alternate heat removal path through the SG(s) is in operation.

This LCO is modified by a Note that prohibits boron dilution with water at a boron concentration less than that required to assure the SDM of LCO 3.1.1 is maintained when SDC forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature, so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the SG(s) can be used as the backup for SDC heat removal. To ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

In MODE 5, it is sometimes necessary to stop all RCP or SDC forced circulation. This is permitted to change operation from one SDC loop to the other, perform surveillance or startup testing, perform the transition to and from the SDC, or to avoid operation below the RCP minimum net positive suction head limit. The time period is acceptable because natural circulation is acceptable for decay heat removal, the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Note 2 allows one SDC loop to be inoperable for a period of up to two hours, provided that the other SDC 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 following conditions be satisfied before an RCP may be started with any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2):

a. Pressurizer water level must be ≤ 170 inches;

b. Pressurizer pressure ≤ 300 psia (Unit 1), ≤ 320 psia (Unit 2); and

c. Secondary side water temperature in each SG must be ≤ 30°F above the RCS temperature. The RCS temperature used for this ΔT evaluation is the average RCS temperature. It may be conservatively measured using the cold leg, SDC return to the RCS, or, more accurately, the average RCS temperature depending on conditions. Where the measurement is taken is controlled by plant procedures.

Satisfying the above conditions will preclude opening a PORV during a pressure transient when the RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting SDC loops to not be in operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the SDC loops.

An OPERABLE SDC loop is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. An OPERABLE SDC loop is supported by a functional saltwater and component cooling water subsystem. Note that one functional saltwater and component cooling water subsystem may support both OPERABLE SDC loops, if its heat removal capacity is sufficient.

SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink when it has an adequate water level and is OPERABLE.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation to remove decay heat from the core and to provide proper boron mixing. One SDC loop provides sufficient circulation for these purposes.

Operation in other MODEs is covered by: LCO 3.4.4, LCO 3.4.5, LCO 3.4.6, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1 and A.2

If the required SDC loop is inoperable and any SGs have secondary side water levels < -50 inches, redundancy for heat removal is lost. Action must be initiated immediately to restore a second SDC loop to OPERABLE status or to restore the water level in the required SGs. Either Required Action A.1 or Required Action A.2 will restore redundant decay heat removal paths. The immediate Completion Times reflect the importance of maintaining the availability of two paths for decay heat removal.

B.1 and B.2

If no SDC loop is in operation, except as permitted in Note 1, all operations involving introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 must be suspended. Action to restore one SDC loop to OPERABLE status and place it in operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. When water is added without forced circulation, unmixed coolant could be introduced to the core, however water added with a boron concentration meeting the minimum SDM maintains an acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that one SDC loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions. In addition, Control Room indication and alarms will normally indicate loop status.

The SDC flow is established to ensure that core outlet temperature is maintained sufficiently below saturation to allow time for swapover to the standby SDC loop should the operating loop be lost.

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are $\geq$ -50 inches ensures that redundant heat removal paths are available if the second SDC loop is inoperable. This surveillance test is required to be performed when the LCO requirement is being met by use of the SGs. If both SDC loops are OPERABLE, this SR is not needed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.7.3

Verification that the second SDC loop is OPERABLE ensures that redundant paths for decay heat removal are available. The requirement also ensures that the additional loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is

BASES

performed by verifying proper breaker alignment and power available to the required pumps and valves that are not in operation. This surveillance test is required to be performed when the LCO requirement is being met by one of two SDC loops, e.g., both SGs have < -50 inches water level. The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES None

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 and transfer of this heat to the SDC heat exchangers. The 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 the SDC System can be used for coolant circulation. 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 SDC loop for decay heat removal and transport and to require that two paths (i.e., SDC loop No. 11 or No. 12 for Unit 1, and SDC loop No. 21 or No. 22 for Unit 2) be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in determining the time available for mitigation of the accidental boron dilution event. The SDC loops provide this circulation. The flow provided by one SDC loop is adequate for decay heat removal and for boron mixing.

Reactor Coolant System loops - MODE 5 (loops not filled) satisfy 10 CFR 50.36(c)(2)(ii), Criterion 4.

LCO

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

Note 1 permits the SDC pumps to not be in operation for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both SDC pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained at least 10°F below saturation temperature. The Note prohibits boron dilution with water at a boron concentration less than that required to assure the SDM of LCO 3.1.1 is maintained or draining operations when SDC forced flow is stopped.

Note 2 allows one SDC loop to be inoperable for a period of two 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.

An OPERABLE SDC loop is composed of an OPERABLE SDC pump capable of providing forced flow to an OPERABLE SDC heat exchanger, along with the appropriate flow and temperature instrumentation for control, protection, and indication. An OPERABLE SDC loop is supported by a functional saltwater and component cooling water subsystem. Note that one functional saltwater and component cooling water subsystem may support both OPERABLE SDC loops, if its heat removal capacity is sufficient. Shutdown cooling pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

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

Operation in other MODEs is covered by: LCO 3.4.4, LCO 3.4.5, LCO 3.4.6, LCO 3.4.7, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1

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

B.1 and B.2

If no SDC loop is OPERABLE or in operation, except as provided in Note 1, all operations involving introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 must be suspended. Action to restore one SDC loop to OPERABLE status and place it in operation must be initiated immediately. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. When water is added without forced circulation, unmixed coolant could be introduced to the core, however water added with a boron concentration meeting the minimum SDM maintains an acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification every 12 hours that one SDC loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions.

SR 3.4.8.2

Verification that the required number of loops are OPERABLE ensures that redundant paths for heat removal are available and that additional loops 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 indicated power available to the required pumps and valves that are not in operation. The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None

B 3.4 REACTOR COOLANT SYSTEMS (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, the required heaters and their backup heater controls, and emergency power supplies. Pressurizer safety valves and pressurizer PORVs are addressed by LCO 3.4.10 and LCO 3.4.11, respectively.

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control, using the sprays and heaters during normal operation and proper pressure response for anticipated design basis transients. The water level limit serves two purposes:

a. Pressure control during normal operation maintains subcooled reactor coolant in the loops and thus in the preferred state for heat transport; and

b. By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes that could result in degraded ability for pressure control.

The maximum water level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus, both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus ensuring that pressure relief devices (PORVs or pressurizer safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the SL of 2750 psia.

The requirement to have two banks of pressurizer heaters, which are permanently powered by Class 1E power supplies, ensures that RCS pressure can be maintained. The pressurizer heaters maintain RCS pressure to keep the reactor coolant subcooled. Inability to control RCS pressure during natural circulation flow could result in loss of single phase flow and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODEs 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. 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 noncondensable gases normally present.

Safety analyses presented in the UFSAR 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.

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 Reference 1, is the reason for their inclusion. The requirement for emergency power supplies is based on Reference 1. The intent is to keep the reactor coolant in a subcooled condition using natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power. While loss of offsite power is a coincident occurrence assumed in the accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated in the accident analyses.

The pressurizer satisfies 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

The LCO requirement for the pressurizer to be OPERABLE with water level ≥ 133 inches and ≤ 225 inches ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two banks of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW and capable of being powered from an emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure. By maintaining the pressure near the operating conditions, a wide subcooling margin to saturation can be obtained in the loops. The generic value of 150 kW is derived from the use of 12 heaters rated at 12.5 kW each. The amount needed to maintain pressure is dependent on the ambient heat losses.

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 RCP startup. The LCO does not apply to MODE 5 (Loops Filled) because LCO 3.4.12 applies. The LCO does not apply to MODEs 5 and 6 with partial loop operation.

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

ACTIONS

A.1 and A.2

With pressurizer water level not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within 12 hours. This takes the plant out of the applicable MODEs and restores the plant to operation within the bounds of the safety analyses. Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Further P/T reduction to MODE 4 brings the plant to a MODE where the LCO is not applicable. The 12 hour time to reach the nonapplicable MODE is reasonable based on operating experience for that evolution.

B.1

If one required bank of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station powered heaters.

C.1 and C.2

If one required bank of pressurizer heaters is inoperable and cannot be restored within 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 Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours is reasonable, based on operating experience to reach MODE 4 from full power to an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR ensures that during steady-state operation, pressurizer water level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The surveillance test is performed by observing the indicated level. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

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 pressurizer heaters are verified to be at their design rating. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Frequency of 24 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

REFERENCES

1. NUREG-0737, II.E.3.1, "Clarification of TMI Action Plan Requirements," November 1980

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the RPS, two valves are used to ensure that the SL of 2750 psia is not exceeded for analyzed transients during operation in MODEs 1 and 2. Two safety valves are used for portions of MODE 3. For the remainder of MODEs 3, 4, 5, and 6 with the head on, overpressure protection is provided by operating procedures and LCO 3.4.12.

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in Reference 1, Section III. The required lift pressures are 2500 psia ± 1% and 2565 psia ± 1%. The safety valves discharge steam from the pressurizer to a quench tank located in the Containment Structure. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The upper and lower pressure limits are based on the ± 1%-tolerance requirement (Reference 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with normal operating pressure. 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. The consequences of exceeding the ASME pressure limit (Reference 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 analyses in the UFSAR that require safety valve actuation, assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of both safety valves and assumes that the valves open at the high range of the as found setting. These valves must accommodate pressurizer insurges that could occur during a loss of load, loss of main feedwater, or main feedwater line break accident. The startup accident establishes the minimum safety valve capacity. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this specification is required to ensure that the accident analysis and design basis calculations remain valid.

The pressurizer safety valves satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

One pressurizer safety valve is set to open at 2500 psia and one is set to open at 2565 psia. These setpoints are within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions, and to comply with ASME Code requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Reference 1) for lifting pressures above 1000 psig. The limit protected by this specification is the RCPB SL of 110% of design pressure. Inoperability of one or both 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 and 2, and portions of MODE 3 above the LTOP temperature, OPERABILITY of two 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 is conservatively included, although the listed accidents may not require both safety valves for protection.

The LCO is not applicable in MODE 3 when all RCS cold leg temperatures are ≤ 365°F (Unit 1), ≤ 301°F (Unit 2), and MODEs 4 and 5, and MODE 6 with the reactor vessel head on, because LTOP is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head off.

The Note allows entry into MODE 3 > 365°F (Unit 1), > 301°F (Unit 2) with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high P/T 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 36 hour exception is based on 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed within 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 RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if two 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 at or below 365°F (Unit 1), 301°F (Unit 2) with all RCS cold leg temperatures ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) within 12 hours. The six hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power without challenging plant systems. Similarly, the 12 hours allowed is reasonable, based on operating experience, to reduce temperature to below 365°F (Unit 1), 301°F (Unit 2) without challenging plant systems. At or below 365°F (Unit 1), 301°F (Unit 2), overpressure protection is provided by LTOP. The change from MODEs 1 or 2, or MODE 3 > 365°F (Unit 1), > 301°F (Unit 2) to MODE 3 ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Surveillance Requirements are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Reference 1, which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valves' setpoints are 2500 psia (+ 2%, - 1%) and 2565 psia (± 2%) for OPERABILITY; however, the valves are reset to ± 1% during the surveillance test to allow for drift.

REFERENCES

1. 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 PORV is an electric, solenoid-operated valve that is automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. The PORV may also be manually opened or closed using a handswitch installed in the Control Room.

An electric, motor-operated, normally open, block valve is installed between the pressurizer and the PORV. The function of the block valve is to isolate the PORV. Block valve closure is accomplished manually using controls in the Control Room and may be used to isolate a leaking PORV to permit continued power operation. Most importantly, the block valve is used to isolate a stuck open PORV to isolate the resulting small break LOCA. Closure terminates the RCS depressurization and coolant inventory loss.

The PORV and its block valve controls are powered from normal power supplies. Their controls are also capable of being powered from emergency supplies. Power supplies for the PORV are separate from those for the block valve. Power supply requirements are defined in Reference 1.

The PORV setpoint is equal to the high pressure reactor trip setpoint and below the opening setpoint for the pressurizer safety valves as required by Reference 2. The purpose of the relationship of these setpoints is to reduce the frequency of challenges to the safety valves, which, unlike the PORV, cannot be isolated if they were to fail open.

The primary purpose of this LCO is to ensure that the PORV and the block valve are operating correctly so the potential for a small break LOCA through the PORV pathway is minimized; or if a small break LOCA were to occur through a failed open PORV, the block valve could be manually operated to isolate the path.

The PORV may be manually-operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available, a condition that may be encountered during loss of offsite power. Operators can manually open the PORVs to reduce RCS pressure in the event of a steam generator tube rupture (SGTR) with offsite power unavailable.

The PORV may also be used for once through 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 PORV functions as an automatic overpressure device and limits challenges to the safety valves. Although the PORV acts as an overpressure device for operational purposes, safety analyses do not take credit for PORV actuation, but do take credit for the safety valves.

The PORV also provides LTOP during heatup and cooldown. Limiting Condition for Operation 3.4.12, addresses this function.

APPLICABLE SAFETY ANALYSES

The PORV small break LOCA break size is bounded by the spectrum of piping breaks analyzed for plant licensing. Because the PORV small break LOCA is located at the top of the pressurizer, the RCS response characteristics are different from RCS loop piping breaks; analyses have been performed to investigate these characteristics.

The possibility of a small break LOCA through the PORV is reduced when the PORV flow path is OPERABLE. The possibility is minimized if the flow path is isolated.

Overpressure protection is provided by safety valves, and analyses do not take credit for the PORV opening for accident mitigation.

Pressurizer PORVs satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The LCO requires the two PORVs and their associated block valves to be OPERABLE. The block valve is required to be

OPERABLE so it may be used to isolate the flow path if the PORV is not OPERABLE.

Valve OPERABILITY also means the PORV setpoint is correct. Ensuring the PORV opening setpoint is correct reduces the frequency of challenges to the safety valves, which, unlike the PORVs, cannot be isolated if they were to fail open.

APPLICABILITY

In MODEs 1 and 2, and MODE 3 with all RCS cold leg temperatures > 365°F (Unit 1), > 301°F (Unit 2), the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. A likely cause for PORV small break LOCA is a result of pressure increase transients that cause 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. Pressure increase transients can occur any time the SGs are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODEs 1 and 2.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, this LCO is applicable in MODEs 1 and 2, and MODE 3 with all RCS cold leg temperatures > 365°F (Unit 1), > 301°F (Unit 2). The LCO is not applicable in MODE 3 with all RCS cold leg temperatures ≤ 365°F (Unit 1), ≤ 301°F (Unit 2), when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODE 3 with T_{avg} ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) and in MODEs 4, 5, and 6 with the reactor vessel head in place. Limiting Condition for Operation 3.4.12 addresses the PORV requirements in these MODEs.

ACTIONS

The ACTIONS are modified by a Note. The Note clarifies that the pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

A.1

With one or two PORVs inoperable and capable of being manually cycled, either the inoperable PORV(s) must be restored or the flow path isolated within one hour. The block valve should be closed but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. Although the PORV may be designated inoperable, it may be able to be manually opened and closed, and in this manner can be used to perform its function. Power-operated relief valve inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use, and do not create a possibility for a small break LOCA. 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. The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE 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 one hour is based on plant operating experience that minor problems can be corrected or closure can be accomplished in this time period.

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

If one PORV is inoperable and not capable of being manually cycled, it must either be isolated, by closing the associated block valve and removing the power from the block valve, or restored to OPERABLE status. The Completion Time of one hour is reasonable, based on challenges to the PORVs during this time period, and provides the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, five days are provided to restore the inoperable PORV to OPERABLE status.

C.1 and C.2

If one block valve is inoperable, then it must be restored to OPERABLE status, or the associated PORV placed in override closed. 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 one hour, the Required Action is to place the PORV in override closed 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 Times of one hour are reasonable based on the small potential for challenges to the system during this time period and provide the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of five days to restore the inoperable block valve to OPERABLE status. 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 are not capable of automatically mitigating an overpressure event when placed in override closed. If the block valve is restored within the Completion Time of five days, the power will be restored and the PORV restored to OPERABLE status.

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

If both PORVs are inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of one hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of one 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 Required Actions D.1 and D.2 have been completed, Required Action D.3 allows 72 hours to restore a PORV to OPERABLE status. This time is reasonable to perform required repairs. This time also accounts for the overpressure protection provided by the pressurizer safety valves in LCO 3.4.10.

E.1 and E.2

If both block valves are inoperable, it is necessary to either restore the block valves within the Completion Time of one hour or place the associated PORVs in override closed and restore at least one block valve to OPERABLE status within 72 hours, and the remaining block valve in five days, per Required Action C.2. The Completion Time of one hour to either restore the block valves or place the associated PORVs in override closed is reasonable based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, then the plant must be brought to a MODE in which the LCO does not apply. The plant must be brought to at least MODE 3 within 6 hours and reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) within 12 hours. The Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours to reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) is reasonable considering that a plant can cool down within that time frame. In MODE 3 with any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) and in MODEs 4, 5, and 6, maintaining PORV OPERABILITY is required per LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

A CHANNEL FUNCTIONAL TEST is performed on each PORV instrument channel every 92 days to ensure the entire channel will perform its intended function when needed.

SR 3.4.11.2

Block valve cycling verifies that it can be closed if necessary. The basis for the Frequency of 92 days is found in Reference 3. If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance because opening the block valve is necessary to permit the PORV to be used for manual control of RCS pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 120 hours, which is well within the allowable limits (25%) to extend the block valve surveillance interval of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Action fulfills the SR).

The Note modifies this SR by stating that this SR is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO.

SR 3.4.11.3

Surveillance Requirement 3.4.11.3 requires complete cycling of each PORV. Power-operated relief valve cycling demonstrates its function. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.

SR 3.4.11.4

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 24 months to adjust the whole channel so that it responds, and the valve opens within the required range and with accuracy to known input.

The 24 month Frequency considers operating experience with equipment reliability and matches the refueling outage Frequency.

REFERENCES

1. NUREG-0737, Paragraph II, G.I, "Clarification of TMI Action Plan Requirements," November 1980
2. Inspection and Enforcement Bulletin 79-05B, "Nuclear Incident at Three Mile Island - Supplement," April 21, 1979
3. ASME Code for Operation and Maintenance of Nuclear Power Plants

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the P/T limits of Reference 1, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. Limiting Condition for Operation 3.4.3, provides the allowable combinations for operational P/T during cooldown, shutdown, and heatup to keep from violating the Reference 1, Appendix G requirements during the LTOP MODEs.

The reactor vessel material is less tough at low temperatures than at normal operating temperatures. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Reference 2). Reactor Coolant System 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. Limiting Condition for Operation 3.4.3 requires administrative control of RCS P/T during heatup and cooldown to prevent exceeding the P/T 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 all but one high pressure safety injection (HPSI) pump incapable of injection into the RCS and this HPSI pump will only be capable of manually injecting into the RCS. When suction of this HPSI pump is aligned to the Refueling Water Tank (RWT), the HPSI pump will be throttled unless an adequate vent path exists. The HPSI motor-operator valves must be in pull-to-override so that valves do not automatically actuate. In addition, administrative controls are placed on charging pump operation. The pressure relief capacity requires either two OPERABLE redundant PORVs, one PORV and an RCS vent of 1.3 square inches, or the RCS depressurized and an RCS vent of 2.6 square inches. One PORV or the 1.3 square inch RCS vent, is the overpressure protection device that acts to terminate an increasing pressure event. The extra PORV or extra 1.3 square inch vent is for single failure criteria.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The safety injection actuation circuits are blocked to HPSI. If conditions require the use of more than one HPSI for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

The LTOP System for pressure relief consists of: two PORVs with reduced lift settings, one PORV with reduced lift setting and an RCS vent of 1.3 square inches, or an RCS vent of 2.6 square inches. Two relief valves are required for redundancy. One PORV has adequate relieving capability to prevent overpressurization for the required coolant input capability.

<u>PORV Requirements</u>

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The actuation logic monitors RCS temperature and pressure, and determines when the LTOP overpressure setting is approached. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The LCO presents the PORV setpoints for LTOP. Having the setpoints of both valves within the limits of the LCO ensures the P/T limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases coolant, the system pressure decreases until a reset pressure is reached. At this point the event is terminated and the operator manually closes the PORV.

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 LTOP 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.

If the vent path is ≥ 8 square inches (e.g., removing the pressurizer manway) the RCS can not be pressurized above the P/T limits, and the LTOP System is not required. A vent path of greater than or equal to 8 square inches also exists during the RCS vacuum fill process when the ≥ 8 square inch vent is temporarily covered with a passive gravity-activated plate that does not obstruct the required flow path when RCS vacuum is lost.

APPLICABLE SAFETY ANALYSES

Safety analyses (Reference 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1, Appendix G, P/T limits during shutdown. In MODEs 1 and 2, and MODE 3 with all RCS cold leg temperatures > 365°F (Unit 1), > 301°F (Unit 2), the RCPB is sufficiently above the nil-ductility temperature that the pressurizer safety valves prevent brittle fracture. At 365°F (Unit 1), 301°F (Unit 2) and below, overpressure prevention falls to the OPERABLE PORVs and administrative controls or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

Each time the P/T limit curves are revised, the LTOP System will be re-evaluated to ensure its functional requirements can still be satisfied using the PORV method or the depressurized and vented RCS condition.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

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

Mass Input Type Transients

a. Inadvertent HPSI pump start;

b. Inadvertent HPSI and charging pump start; or

c. Charging/letdown flow mismatch.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of SDC; or

c. Reactor coolant pump startup with temperature asymmetry within the RCS or between the RCS and SGs.

The following are required during the LTOP MODEs to ensure that mass and heat input transients do not occur which either of the LTOP overpressure protection means cannot handle:

a. Rendering all but one HPSI pump incapable of injection and blocking automatic initiation from the remaining HPSI pump;

b. When HPSI suction is aligned to the RWT, the HPSI pump shall be in manual control and either:

   1) HPSI flow is limited to ≤ 210 gpm, or

   2) an RCS vent > 2.6 square inches is established;

c. Rendering HPSI motor operated valves (MOVs) only capable of manually aligning HPSI pump flow to the RCS;

d. Running only one charging pump when injecting via HPSI (charging pump requirements are controlled administratively); and

e. Maintaining a pressure bubble with level ≤ 170 inches.

The Reference 3 analyses demonstrate that either one PORV or the RCS vent and pressurizer steam volume can maintain RCS pressure below limits when only one HPSI pump is actuated and the HPSI pump's flow is throttled. If HPSI pump flow is not throttled during addition of mass to the RCS through on HPSI loop MOV, then two PORVs or an RCS vent $\geq$ 2.6 square inches are capable of maintaining RCS pressure below limits. Thus, the LCO allows only one HPSI pump OPERABLE with flow throttled, or with an RCS vent $\geq$ 2.6 square inches during the LTOP MODEs.

Also to limit pressure overshoot over the PORV setpoint, the remaining HPSI and two charging pumps are rendered incapable of injection, and the RCPs are disabled during water solid operation.

Heatup and cooldown analyses established the temperature of LTOP Applicability at 365°F (Unit 1), and 301°F (Unit 2) and below, based on Standard Review Plan criteria. Above this temperature, the RCPB is sufficiently above the nil-ductility temperature and the pressurizer safety valves provide the reactor vessel pressure protection against brittle fracture. The vessel materials were assumed to have a fluence level equal to 4.49×10^{19} n/cm^2 (Unit 1), 4.0×10^{19} n/cm^2 (Unit 2).

The consequences of a LOCA in LTOP conform to Reference 1, Appendix K and 10 CFR 50.46, requirements, by having SITs operable in MODE 3 and one HPSI pump available for manual actuation.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the curves in Figure 3.4.12-1 and are applicable when the SDC System is not in operation. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting case of loss of SDC and one charging pump injecting into the RCS during water solid operation. These analyses consider pressure overshoot beyond the PORV opening setpoints, resulting from signal processing and valve stroke times. The PORV setpoints below the derived limit ensure the Reference 1, Appendix G limits will be met. When the SDC System is in operation, the PORV lift setting must be

≤ 429 psia (Unit 1), ≤ 443 psia (Unit 2). This ensures that the PORV lift setting is low enough to mitigate overpressure transients when SDC is in operation, since RCS temperature measurement is not accurate in this condition.

The PORV setpoints will be re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement caused by neutron irradiation. Revised P/T 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 discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV represents the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses shows a vent size of 1.3 square inches is capable of mitigating the limiting allowed LTOP overpressure transient provided a pressurizer steam volume exists, two of the three HPSI pumps are disabled and the remaining HPSI pump's flow is throttled. In that event, this size vent maintains RCS pressure less than the maximum RCS pressure on the P/T limit curve. A 2.6 square inch vent is required to allow for single failures of other equipment, such as HPSI throttle valves. An 8 square inch vent is sufficient to preclude RCS overpressure events. Therefore, when an 8 square inch vent is established, LTOP System requirements are not necessary to maintain RCS pressure within limits.

The RCS vent size will also 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.

LTOP System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

This LCO is required to ensure that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and 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, Appendix G limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires a maximum of one HPSI pump only capable of manually injecting into the RCS. This is accomplished by disabling two HPSI pumps by either removing (racking out) their motor circuit breakers from the electrical power supply circuit or by locking shut their discharge valves. During required testing, other means of preventing two HPSI pumps from injecting into the RCS may be used. In addition, when not in use the remaining HPSI pump shall have its handswitch in pull-to-lock. When HPSI suction is aligned to the RWT for injection into the RCS, the HPSI pump must be in manual control, and either HPSI flow shall be limited to ≤ 210 gpm, or an RCS vent of ≥ 2.6 square inches is established. To provide single failure protection against a HPSI pump mass addition transient, the HPSI loop MOV handswitches must be placed in pull-to-override so the valves do not automatically actuate upon receipt of a safety injection signal. During required testing this requirement may be suspended.

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

a. Two OPERABLE PORVs and associated block valves open;

b. One OPERABLE PORV and associated block valve open and an RCS vent open with an area ≥ 1.3 square inches; or

c. The depressurized RCS and an RCS vent open with an area ≥ 2.6 square inches.

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set in accordance with the LCO and testing has proven its ability to open at that setpoint, and motive power is available to the two valves and their control circuits.

The combination of these methods of overpressure prevention (as specified in LCO 3.4.12) are capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 3 when the temperature of any RCS cold leg is ≤ 365°F (Unit 1), ≤ 301°F (Unit 2), in MODEs 4, 5, and 6.

Limiting Condition for Operation 3.4.3 provides the operational P/T limits for all MODEs. Limiting Condition for Operation 3.4.10, requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODEs 1 and 2, and MODE 3 above 365°F (Unit 1), 301°F (Unit 2).

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.

The Applicability is modified by a Note stating that this Specification is not applicable when the RCS is vented ≥ 8 square inches. An RCS vent of this size precludes RCS overpressure events.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to inoperable PORVs used for LTOP. There is an increased risk associated with entering MODE 3 from MODE 4 with PORVs used for LTOP 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

With one or more HPSI pumps capable of automatically injecting into the RCS or with two or more HPSI pumps capable of manually injecting into the RCS, overpressurization is possible.

The immediate Completion Time to initiate actions to restore restricted coolant input capability to the RCS reflects the importance of maintaining overpressure protection of the RCS.

B.1

With HPSI flow > 210 gpm and suction aligned to the RWT and an RCS vent < 2.6 square inches established, sufficient overpressure protection may not exist and overpressurization may be possible.

The immediate Completion Time to initiate actions to reduce HPSI flow to ≤ 210 gpm reflects the importance of maintaining overpressure protection of the RCS.

C.1

With one or more HPSI loop MOVs capable of automatically aligning HPSI pump flow to the RCS, single failure protection against a HPSI pump mass addition transient is lost. Therefore, action is required to be immediately initiated to restore single failure protection by placing the affected HPSI loop MOV handswitch to pull-to-override, or shutting and disabling the affected HPSI loop MOV, or isolating the affected HPSI header flow path.

The immediate Completion Time to initiate action to restore single failure protection for the HPSI pump mass addition transient reflects the importance of restoring single failure protection for low temperature overpressurization mitigation.

D.1

In MODE 3 when any RCS cold leg temperature is ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) or in MODE 4, with one of the two required PORVs inoperable and an RCS vent < 1.3 square inches established, the inoperable PORV must be restored to OPERABLE status within a Completion Time of five days. The inoperable PORV is required to meet the LCO requirement and to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time is based on the fact that only one PORV is required to mitigate an overpressure transient.

E.1

The consequences of operational events that will overpressure the RCS are more severe at lower temperature (Reference 4). Thus, with one of the two required PORVs inoperable and an RCS vent < 1.3 square inches established in MODE 5 or in MODE 6, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The 24 hour Completion Time to restore the inoperable PORV to OPERABLE status in MODE 5 or in MODE 6 is a reasonable amount of time to investigate and repair several types of PORV failures without exposure to a lengthy period with only one PORV OPERABLE to protect against overpressure events.

F.1

If the required Actions and associated Completion Times of Conditions D or E cannot be met, the RCS is required to be depressurized and vented through a vent ≥ 1.3 square inches. This action must be completed within 48 hours. This action along with the OPERABLE PORV restores single failure protection and ensures the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODEs. This action protects the RCPB from an overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time of 48 hours to depressurize and vent the RCS is based on the time required to place the plant in this condition and in a controlled manner. The probability of an overpressure event occurring along with a single failure of the remaining OPERABLE PORV is unlikely.

G.1

If all required PORVs (i.e., when one PORV is required and it is inoperable or when two PORVs are required and both are inoperable) are inoperable, the RCS must be depressurized and a vent established within 48 hours. The vent must be sized at least 2.6 square inches to ensure the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODEs. This action protects the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time of 48 hours to depressurize and vent the RCS is based on 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 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, verification that a maximum of one HPSI pump is only capable of manually injecting into the RCS, and automatic alignment of the HPSI loop MOVs, is prevented (by disabling the automatic opening features of the HPSI loop MOVs) is required. The HPSI pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control or by verifying their discharge valves are locked shut.

The 12 hour interval considers operating practice to regularly assess potential degradation and to verify operation within the safety analysis.

SR 3.4.12.3

Surveillance Requirement 3.4.12.3 requires verifying that the required RCS vent is open, once every 12 hours for a valve that is unlocked open, and once every 31 days for a valve that is locked open.

The passive vent arrangement must only be open to be OPERABLE. This SR need only be performed if the vent is being used to satisfy the requirements of this LCO. The Frequencies consider operating experience with mispositioning of unlocked and locked vent valves, respectively.

SR 3.4.12.4

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve can be remotely verified open in the main Control Room.

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

The 72 hour Frequency considers operating experience with accidental movement of valves having remote control and position indication capabilities available where easily monitored. These considerations include the administrative controls over main Control Room access and equipment control.

SR 3.4.12.5

Performance of a CHANNEL FUNCTIONAL TEST is required every 31 days to verify and, as necessary, adjust the PORV open setpoints. The CHANNEL FUNCTIONAL TEST will verify on a monthly basis that the PORV lift setpoints are within the LCO limit. 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 FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. Power-operated relief valve actuation could depressurize the RCS and is not required. The 31 day Frequency considers experience with equipment reliability.

A Note has been added indicating this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to ≤ 365°F (Unit 1), ≤ 301°F (Unit 2). The test cannot be performed until the RCS is in the LTOP MODEs when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODEs.

SR 3.4.12.6

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 24 months to adjust the whole channel so that it responds and the valve opens within the required LTOP range and with accuracy to known input.

The 24 month Frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.

REFERENCES

1. 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities
2. Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations, July 12, 1988
3. UFSAR, Section 4.2.2, Low Temperature Overpressure Protection
4. Generic Letter 90-06, Resolution of Generic Issues 70, "PORV and Block Valve Reliability," and 94, "Additional LTOP Protection for PWRs," June 25, 1990

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 makeup the RCS. Component joints are made by welding, bolting, rolling, or pressure loading. Valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce 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 LEAKAGE.

Reference 1, Appendix 1C, Criterion 16 requires means for detecting reactor coolant LEAKAGE. Reference 2 describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS 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 detrimental to the safety of the facility and the public.

A limited amount of leakage inside Containment Structure 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 RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a LOCA.

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 100 gpd/SG primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a major factor in the dose releases outside the Containment Structure resulting from a steam line break accident. To a lesser extent, other accidents, or transients such as, a SGTR, a seized rotor event, and a Control Element Assembly Ejection event, involve secondary contamination via primary to secondary leakage and subsequent secondary steam release to the atmosphere via the atmospheric dump valves and main steam safety valves.

Although the operational leakage rate limit applies to leakage through any one SG, Reference 5 requires that the leakage be apportioned between steam generators in such a manner that the calculated dose is maximized. The analyses described in Reference 1 include appropriate apportioning of steam generator leakage to maximize the dose consequences. Of the four secondary release accidents, the SGTR with a Preaccident Iodine Spike is the most limiting for offsite radiation releases.

Reactor Coolant System operational LEAKAGE satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

Reactor Coolant System 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 degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall is RCS pressure boundary LEAKAGE. Zero leakage in an isolated fault is not pressure boundary LEAKAGE. A fault is considered isolated with no pressure boundary LEAKAGE, if following the positioning of isolation device(s) it is verified that zero leakage exists.

b. Unidentified LEAKAGE

One gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment, 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 the detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the Containment Structure from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled RCP seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One Steam Generator

The limit of 100 gpd per SG is based on a safety analysis assumption. Plant procedures further limit operational LEAKAGE to 50 gpd/SG to ensure the TS operational limit of 100 gpd/SG assumed in the accident analysis will not be exceeded as a result of additional LEAKAGE induced during the accident. This limit is more conservative than the 150 gpd/SG operational LEAKAGE performance criterion in Reference 3. The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage.

The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of SG tube ruptures.

APPLICABILITY

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

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

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within four 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 primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within four hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and to 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 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 LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady-state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, and makeup and letdown). The surveillance 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 required to perform a proper water inventory balance; calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady-state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal leakoff flows.

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. These leakage detection systems are specified in LCO 3.4.14.

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 100 gpd cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 100 gpd 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 3.4.18, "Steam Generator Tube Integrity," should be evaluated. The 100 gpd limit is measured at hot plant conditions as described in Reference 4. 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.

The Surveillance 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 the RCS primary to secondary LEAKAGE determination, steady-state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, and makeup and letdown.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the guidelines in Reference 4.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973
3. Nuclear Energy Institute (NEI) 97-06, Steam Generator Program Guidelines
4. Electric Power Research Institute, Pressurized Water Reactor Primary-to-Secondary Leakage Guidelines
5. 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 Leakage Detection Instrumentation

BASES

BACKGROUND

Reference 1, Appendix 1C, Criterion 16 requires means for detecting RCS LEAKAGE. Reference 2 describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant RCPB degradation, as soon after the 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. In addition to meeting the OPERABILITY requirements, the monitors are typically set to provide the most sensitive response without causing an excessive number of spurious alarms.

The containment sump used to collect unidentified LEAKAGE is instrumented to alarm when level increases above the alarm trip setpoint. The sump is then drained and time logged. If the alarm sounds again, the time is logged and a leakage rate is calculated. This is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the Containment Structure, may be detected by radiation monitoring instrumentation. Radioactivity detection systems are included for monitoring both particulate and gaseous activities, because of their sensitivities and responses to RCS LEAKAGE. These radioactivity monitors have a range of 10^1-10^6 counts per minute.

Other indications may be used to detect an increase in unidentified LEAKAGE; however they are not required to be OPERABLE by this LCO. An increase in humidity of the containment atmosphere would indicate release of water vapor to the Containment Structure, which would be an indicator of potential RCS LEAKAGE. 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. Humidity level monitoring is considered most useful as an 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 Structure. 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 Structure. 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 Structure. Temperature and pressure monitors are not required by this LCO. The above-mentioned LEAKAGE detection methods or systems differ in sensitivity and response time. Some of these systems could serve as early alarm systems signaling the operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required.

APPLICABLE SAFETY ANALYSES

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 RCS leakage detection instrumentation is described in Reference 1, Section 4.3.

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 are necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should leakage occur detrimental to the safety of the facility and the public.

Reactor Coolant System leakage detection instrumentation satisfies 10 CFR 50.36(c)(2)(ii), Criterion 1.

LCO

This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide confidence that small amounts of unidentified LEAKAGE are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requires two instruments to be OPERABLE.

The containment sump is used to collect unidentified LEAKAGE. The monitor on the containment sump detects level. The identification of an increase in unidentified LEAKAGE will be delayed by the time required for the unidentified LEAKAGE to travel to the containment sump and it may take longer than one hour to detect a 1 gpm increase in unidentified LEAKAGE, depending on the origin and magnitude of the LEAKAGE. This sensitivity is acceptable for containment sump level alarm OPERABILITY.

The reactor coolant contains radioactivity that, when released to the containment, may be detected by the gaseous or particulate containment atmosphere radioactivity monitor. Only one of the two detectors is required to be OPERABLE. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE, but have recognized limitations. 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. If there are few fuel element cladding defects and low levels of activation products, it may not be possible for the gaseous or particulate containment atmosphere radioactivity monitors to detect a 1 gpm increase within 1 hour during normal operation. However, the gaseous or particulate containment atmosphere radioactivity monitor is OPERABLE when it is capable of detecting a 1 gpm increase in unidentified LEAKAGE within 1 hour given an RCS activity equivalent to that assumed in the design calculations for the monitors (Reference 1).

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor, in combination with a particulate or gaseous radioactivity monitor, provides an acceptable minimum.

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 MODEs 5 or 6, the temperature is ≤ 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 is much smaller. Therefore, the requirements of this LCO are not applicable in MODEs 5 and 6.

ACTIONS

A.1 and A.2

If the containment sump level alarm is inoperable, no other form of sampling can provide the equivalent information.

However, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the containment atmosphere radioactivity 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 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 sump level alarm to OPERABLE status is required to regain the function in a Completion Time of 30 days after the monitor'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, and B.2

With both gaseous and particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab 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. 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 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. With a sample obtained and analyzed, or an inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors.

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

C.1 and C.2

With the containment sump level alarm inoperable, the only means of detecting LEAKAGE is the required containment atmosphere radiation monitor. A Note clarifies that this Condition is applicable when the only OPERABLE monitor is the containment atmosphere gaseous radiation monitor. The containment atmosphere gaseous radioactivity monitor typically cannot detect a 1 gpm leak within one hour when RCS activity is low. In addition, this configuration does not provide the required diverse means of leakage detection. Indirect methods of monitoring RCS LEAKAGE must be implemented. Grab samples of the containment atmosphere must be taken and analyzed every 12 hours to provide alternate periodic information. The 12 hour interval is sufficient to detect increasing RCS LEAKAGE. The Required Action provides 7 days to restore the containment sump level alarm to OPERABLE status to regain the intended leakage detection diversity. The 7 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

D.1 and D.2

If any required Action of Conditions A, B, or C cannot be met within the required 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 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.

E.1

If all required alarms and monitors are inoperable, no automatic means of monitoring leakage are available, an immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Surveillance Requirement 3.4.14.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.14.2

Surveillance Requirement 3.4.14.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitors. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. 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 FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.14.3 and SR 3.4.14.4

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 Structure. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Operating experience has shown this Frequency is acceptable.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.45, Revision 0, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Specific Activity

BASES

BACKGROUND	Title 10 CFR 50.67 and Reference 2 specify the maximum TEDE an individual at the exclusion area boundary can receive for two hours and that individuals in the low population zone and the Control Room can receive over 30 days during an accident. The limits on specific activity ensure that the doses are held to within the acceptance criteria given in Reference 1, Chapter 14, 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 offsite and Control Room radioactivity dose consequences in the event of a SGTR accident. The LCO limits also impact, to a much lesser extent, other transients such as the maximum hypothetical accident, the seized rotor event, the main steal line break, and the Control Element Assembly Ejection event. The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross activity. The allowable levels are intended to limit the dose at the exclusion area boundary and the Control Room to within the acceptance criteria given in Reference 1, Chapter 14.
APPLICABLE SAFETY ANALYSIS	The LCO limits on the specific activity of the reactor coolant ensure that the resulting doses at the exclusion area boundary or in the Control Room will not exceed the acceptance criteria given in Reference 1, Chapter 14. The SGTR safety analysis (Reference 1, Section 14.15) assumes the specific activity of the reactor coolant at the LCO limits including either a Preaccident Iodine Spike or a Concurrent Iodine Spike and assumes that all of the 200 gpd primary to secondary leakage is to the unaffected SG, since primary to secondary flow through the ruptured SG tube has already been maximized. The rise in pressure in the ruptured SG causes radioactively contaminated steam to discharge to the atmosphere through the atmospheric dump valves and the main steam safety valves.

BASES

The safety analysis shows the radiological consequences of an SGTR accident, are within the Reference 1, Chapter 14 acceptance criteria. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.15-1 for more than 100 hours.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.15-1, are acceptable because of the low probability of an SGTR accident occurring during the established 100 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the exclusion area boundary dose levels beyond the acceptance criteria given in Reference 1, Chapter 14.

Reactor Coolant System specific activity satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The specific activity is limited to 0.5 μCi/gm DOSE EQUIVALENT I-131, and the gross activity in the primary coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limits on DOSE EQUIVALENT I-131 and gross activity ensure the TEDE dose to an individual at the exclusion area boundary during the DBA will be within the acceptance criteria given in Reference 1, Chapter 14. The limit on gross activity ensures the whole body dose to an individual at the exclusion area boundary during the DBA will be within the acceptance criteria given in Reference 1, Chapter 14.

The SGTR accident analysis (Reference 1, Section 14.15) shows that the exclusion area boundary and Control Room dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to exclusion area boundary and Control Room doses that exceed the Reference 1, Chapter 14 acceptance criteria.

APPLICABILITY

In MODEs 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross activity is necessary to contain the potential consequences of an SGTR to within the acceptable exclusion area boundary and Control Room dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODEs 4 and 5, the release of radioactivity in the event of an SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the atmospheric dump valves and main steam safety valves.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of four hours must be taken to demonstrate the limits of Figure 3.4.15-1 are not exceeded. The Completion Time of four hours is required to obtain and analyze a sample.

Sampling must continue for trending. The DOSE EQUIVALENT I-131 must be restored to within limits, within 100 hours. The Completion Time of 100 hours is required if the limit violation resulted from normal iodine spiking.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(s) while relying on the ACTIONS. 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 DOSE EQUIVALENT I-131 specific activity excursions while the plant remains at, or proceeds to power operation.

B.1

If a Required Action and associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.15-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within six hours. The allowed Completion Time of six hours is required to reach MODE 3 below 500°F without challenging plant systems.

C.1

With the gross activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.

The change within six hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of six hours is required to reach MODE 3 below 500°F from full power conditions and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

The SR requires performing a gamma isotopic analysis, as a measure of the gross activity of the reactor coolant, at least once per seven days. While $\bar{E}$ is basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this gamma isotopic measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This SR provides an indication of any increase in gross activity.

Trending the results of this SR allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The SR is applicable in MODEs 1 and 2, and in MODE 3 with RCS average temperature at least 500°F. The seven day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.15.2

This SR is performed to ensure iodine remains within limits during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross activity is monitored every 7 days. The Frequency, between two hours and six hours after a power change of ≥ 15% RTP within a one hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

The SR is modified by a Note which requires the surveillance test to only be performed in MODE 1. This is required because the level of fission products generated in other MODEs is much less. Also, fuel failures associated with fast power changes is more apt to occur in MODE 1 than in MODEs 2 and 3.

SR 3.4.15.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (six months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is not required to be performed until 31 days after 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. This ensures the radioactive materials are at equilibrium so that analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. UFSAR
2. 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 3.4.16 Special Test Exception (STE) RCS Loops - MODE 2

BASES

BACKGROUND

This STE to LCO 3.4.4 and LCO 3.3.1, permits reactor criticality under no flow conditions during PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) while at low THERMAL POWER levels. Reference 1, requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant.

The key objectives of a test program are to: provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the facility has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations.

The tests will include: verifying the ability to establish and maintain natural circulation following a plant trip between 10% and 15% RTP, performing natural circulation cooldown on emergency power, and (during the cooldown), showing that adequate boron mixing occurs and that pressure can be controlled using auxiliary spray and pressurizer heaters powered from the emergency power sources.

APPLICABLE SAFETY ANALYSES

As described in LCO 3.0.7, compliance with Special Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

BASES

LCO

This LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without this LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODEs 1 and 2), and the appropriate tests could not be performed.

In MODE 2, where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER is $<$ 5% RTP and the reactor trip setpoints of the OPERABLE power level channels are set $\leq$ 15% RTP. These limits ensure no SLs or fuel design limits will be violated.

The exception is allowed even though there are no bounding safety analyses. These tests are allowed since they are performed under close supervision during the test program and provide valuable information on the plant's capability to cool down without offsite power available to the RCPs.

APPLICABILITY

This LCO ensures that the plant will not be operated in MODE 1 without forced circulation. It only allows testing under these conditions while in MODE 2. This testing establishes that heat input from nuclear heat does not exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.

ACTIONS

A.1

If THERMAL POWER increases to $>$ 5% RTP, the reactor must be tripped immediately. This ensures the plant is not placed in an unanalyzed condition and prevents exceeding the specified acceptable fuel design limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

THERMAL POWER must be verified to be within limits once per hour to ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The hourly Frequency has been shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation is within the LCO limits. Plant operations are conducted slowly during the performance of PHYSICS TESTS, and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.

SR 3.4.16.2

Within 12 hours of initiating startup or PHYSICS TESTS, a CHANNEL FUNCTIONAL TEST must be performed on each logarithmic power level neutron flux monitoring channel to verify OPERABILITY and adjust setpoints to proper values. This will ensure that the RPS is properly aligned to provide the required degree of core protection during startup or the performance of the PHYSICS TESTS. 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 FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. The interval is adequate to ensure that the appropriate equipment is OPERABLE prior to the tests to aid the monitoring and protection of the plant during these tests.

REFERENCES

1. 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Section XI

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Special Test Exception (STE) RCS Loops - MODES 4 and 5

BASES

BACKGROUND	This STE to LCO 3.4.6, LCO 3.4.7, and LCO 3.4.8, allows no RCS or SDC loops to be in operation during the time intervals required: 1) for local leak rate testing of Containment Penetration Number 41 (SDC); and 2) for maintenance on the common SDC suction line or on the SDC flow control valve (CV-306).
APPLICABLE SAFETY ANALYSIS	As described in LCO 3.0.7, compliance with Special Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) applies. Special Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criterion satisfied for the other LCOs is provided in their respective Bases.
LCO	This LCO is provided to allow for the performance of testing and maintenance in MODEs 4 and 5 (normally after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without this LCO, plant operations would be held bound to the normal operation LCOs for reactor coolant loops and circulation (MODEs 4 and 5), and the appropriate tests or maintenance could not be performed in these MODEs. In MODEs 4 and 5, operation is allowed under no flow conditions provided: the xenon reactivity is $\leq 0.1\%\ \Delta k/k$ and approaching stability, no operations are permitted which could cause introduction of water into the RCS with a boron concentration less than that required by LCO 3.1.1, the charging pumps are de-energized, the charging flow paths are isolated, and the SHUTDOWN MARGIN requirement of LCO 3.1.1 is verified at least once per eight hours. These limits along with the SRs ensure no SLs or fuel design limits will be violated. The exception is allowed even though there are no bounding safety analyses. These tests or maintenance are allowed since they are performed under close supervision during the test program and must stay within the requirements of the LCO.

BASES

APPLICABILITY

The LCO ensures that while within this LCO the plant will not be operated in any other MODE besides MODEs 4 and 5 without forced circulation. This is because the MODEs of Applicability for this Specification are MODEs 4 and 5. This Specification allows testing and maintenance to be performed on the SDC System while SDC is required to be OPERABLE.

ACTIONS

A.1

If one or more requirements of the LCO are not met, all activities being performed under this STE must be immediately suspended. These activities are local leak rate testing of the SDC penetration and maintenance on valves in the SDC System. The Completion Time to suspend these activities immediately ensures the plant is not placed in an unanalyzed condition and prevents exceeding the specified acceptable fuel design limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

Xenon reactivity must be verified to be within limits once within one hour prior to suspending the reactor coolant circulation requirements of LCO 3.4.6, LCO 3.4.7, and LCO 3.4.8. The frequency of once within one hour prior to suspending the applicable RCS Loops LCO will ensure that the xenon reactivity is within limits and trending toward stability prior to suspending forced flow cooling. This will ensure no SLs or fuel design limits will be violated while testing or maintenance are being conducted.

SR 3.4.17.2 and SR 3.4.17.3

Verifying the charging pumps are de-energized and the charging flow paths are isolated, ensures that the major source of a boron reduction is not available. These two SRs support the requirement that no source be available that could cause an RCS boron concentration reduction. These SRs are required to be verified at a frequency of one hour. The one hour frequency is sufficient to ensure that these sources will not be available to cause a reduction of the RCS boron concentration.

Subsequent performance of these SRs after the initial verification that the charging pumps are de-energized and the charging flow paths are isolated, may be performed administratively.

SR 3.4.17.4

This SR requires that a SHUTDOWN MARGIN verification be performed in accordance with SR 3.1.1.1 once per eight hours. The normal Frequency for these SRs is once per 24 hours. The eight hour Frequency reflects that no forced flow cooling is available and that the SHUTDOWN MARGIN should be verified more frequently. The eight hour Frequency is sufficient to ensure that the SHUTDOWN MARGIN remains within limits while under this STE.

REFERENCES None

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam Generator 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 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, LCO 3.4.5, LCO 3.4.6, and LCO 3.4.7.

Steam generator 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, 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 Reference 1.

APPLICABLE SAFETY ANALYSIS

The SGTR event 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 bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, plus the leakage rate associated with a double-ended rupture of a single tube. Reference 7 requires that the leakage be apportioned between SGs in such a manner that the calculated dose is maximized. For the SGTR, all 200 gpd primary to secondary leakage is assumed to be to the unaffected SG, since primary to secondary flow through the ruptured SG tube has already been maximized. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via atmospheric dump valves and main steam safety valves.

The analysis for design basis accidents and transients other than a SGTR assume 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 100 gpd/SG or is assumed to increase to 100 gpd/SG as a result of accident induced conditions. Reference 7 requires that the leakage should be apportioned between SGs in such a manner that the calculated dose is maximized. 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 3.4.15 limits assuming the relevant Iodine spiking factors. 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 General Design Criteria (GDC) 19 (Reference 2), 10 CFR 50.67 (Reference 3), and the NRC accident dose criteria of Reference 7.

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 repair criteria be plugged in accordance with the Steam Generator Program.

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

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

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

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

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

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions), and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on References 4 and 5.

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 the total accident leakage does not exceed 100 gpd/SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident. Reference 7 requires that the leakage should be apportioned between SGs in such a manner that the calculated dose is maximized.

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 and limits primary to secondary LEAKAGE through any one SG to 100 gpd. 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.

Reactor Coolant System 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.

<u>A.1 and A.2</u>

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.18.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair 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.

SURVEILLANCE REQUIREMENTS

SR 3.4.18.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. Reference 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 repair 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 technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.18.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Reference 6). The Steam Generator Program uses information on existing degradation 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.

SR 3.4.18.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.5.9 are intended to 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 repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met util the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessment 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 repair 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. 10 CFR Part 50, Appendix A, GDC 19
3. 10 CFR 50.67
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB
5. Draft Regulatory Guide 1.121, Basis for Plugging Degraded Steam Generator Tubes, August 1976
6. EPRI, Pressurized Water Reactor Steam Generator Examination Guidelines
7. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000