B 3.3 INSTRUMENTATION

B 3.3.8 Alternate Shutdown System

BASES

BACKGROUND

The Alternate Shutdown System provides the control room operator with sufficient instrumentation and controls to maintain the plant in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the plant in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator safety valves or the steam generator atmospheric dump valves can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Primary Coolant System (PCS) from outside the control room allow extended operation in MODE 3.

The Auxiliary Hot Shutdown Panels (C-150/C-150A) are located in the southwest electrical penetration room. These panels are comprised of two enclosures, the main enclosure C-150 and an auxiliary enclosure C-150A. The description below combines these two enclosures into one entity "Panel C-150."

Panel C-150 provides control of the AFW flow control valves and AFW turbine steam supply Valve. Indication of AFW flow, Steam Generator water level , pressurizer pressure, and pressurizer level are provided. See FSAR Section 7.4 (Ref. 1) for operation via Panel C-150.

The instrumentation and equipment controls that are required are listed in Table 3.3.8-1.

Switches, which transfer control or instrument functions from the control room to the C-150 panel, alarm in the control room when the C-150 panel is selected.

APPLICABLE SAFETY ANALYSES

The Alternate Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to maintain the plant in a safe condition in MODE 3.

The criteria governing the design and the specific system requirements of the Alternate Shutdown System are located in 10 CFR 50, Appendix A, GDC 19, and Appendix R (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Alternate Shutdown System has been identified as an important contributor to the reduction of plant risk to accidents and, therefore, satisfies the requirements of Criterion 4 of 10 CFR 50.36(c)(2).

LCO

The Alternate Shutdown System LCO provides the requirements for the OPERABILITY of one channel of the instrumentation and controls necessary to maintain the plant in MODE 3 from a location other than the control room. The instrumentation and controls required are listed in Table 3.3.8-1 in the accompanying LCO.

Equipment controls that are required by the alternative dedicated method of maintaining MODE 3 are as follows:

1. AFW flow control valves (CV-0727 and CV-0749); and
2. Turbine-driven AFW pump.

Instrumentation systems displayed on the Auxiliary Hot Shutdown Control Panel are:

1. Source range flux monitor;
2. AFW flow (HIC-0727 and HIC-0749C);
3. Pressurizer pressure;
4. Pressurizer level;
5. SG level and pressure;
6. Primary coolant temperatures (hot and cold legs);
7. Turbine-driven AFW pump low-suction pressure warning light; and
8. SIRW tank level.

A Function of an Alternate Shutdown System is OPERABLE if all instrument and control channels needed to support the remote shutdown Functions are OPERABLE.

BASES

LCO (continued)

The Alternate Shutdown System instrumentation and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure that the instrument and control circuits will be OPERABLE if plant conditions require that the Alternate Shutdown System be placed in operation.

Table 3.3.8-1 Indication Channel 1, Source Range Nuclear Instrumentation, uses the same detector and preamplifier as the control room channel. Optical isolation is provided between the control room and AHSDP (Alternate Hot Shut Down Panel) portions of the circuit. When the control switches are changed to the "AHSDP" position, the detector and preamplifier is isolated from its normal power supply and connected into the AHSDP power supply.

Table 3.3.8-1 Indication Channels 2 and 12 are provided with their own pressure and level transmitter. The associated circuitry is energized when the AHSDP is energized.

The other Table 3.3.8-1 Indication Channels in Table 3.3.8-1 use a transmitter which also serves normal control room instrumentation. When the control switches are changed to the "AHSDP" (Alternate Hot Shut Down Panel) position, the transmitter is isolated from its normal power supply and circuitry, and connected into the C-150 or C-150A panel circuit; control for AFW flow control valves CV-0727 and CV-0749 is also transferred to C-150. The transfer switches are alarmed in the control room.

APPLICABILITY

The Alternate Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the plant can be maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the plant is already subcritical and in the condition of reduced PCS energy. Under these conditions, considerable time is available to restore necessary instrument control Functions if control room instruments or control become unavailable.

BASES

ACTIONS

A Note has been included that excludes the MODE change restrictions of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS, even though the ACTIONS may eventually require a plant shutdown. This is acceptable due to the low probability of an event requiring this system. The Alternate Shutdown System equipment can generally be repaired during operation without significant risk of spurious trip.

Note 2 has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1. The Completion Time of the inoperable channel of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

A.1

Condition A addresses the situation where the required channels of the Remote Shutdown System are inoperable. This includes any Function listed in Table 3.3.8-1 as well as the control and transfer switches.

Required Action A.1 is to restore the channel to OPERABLE status within 30 days. This allows time to complete repairs on the failed channel. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A are not met, 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 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

This SR applies to the startup range neutron flux monitoring channel. The CHANNEL FUNCTIONAL TEST consists of verifying proper response of the channel to the internal test signals, and verification that a detectable signal is available from the detector. After lengthy shutdown periods flux may be below the range of the channel indication. Signal verification with test equipment is acceptable.

The CHANNEL FUNCTIONAL TEST of the startup range neutron flux monitoring channel is performed once within 7 days prior to reactor startup. The Frequency is based on plant operating experience that demonstrates channel failure is rare.

SR 3.3.8.2

SR 3.3.8.2 verifies that each required Alternate Shutdown System transfer switch and control circuit performs its intended function. This verification is performed from AHSDPs C-150 and C-150A and locally, as appropriate. Operation of the equipment from the AHSDPs C-150 and C-150A is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the plant can be maintained in MODE 3 from the auxiliary shutdown panel and the local control stations.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience demonstrates that Alternate Shutdown System control channels seldom fail to pass the Surveillance when performed at a Frequency of once every 18 months.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.3

A CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to the measured parameter within the necessary range and accuracy.

Performance of a CHANNEL CALIBRATION every 18 months on Functions 1 through 15 ensures that the channels are operating accurately and within specified tolerances. This verification is performed from the AHSDPs and locally, as appropriate. A test of the AFW pump suction pressure alarm (Function 15) is included as part of its CHANNEL CALIBRATION. This will ensure that if the control room becomes inaccessible, the plant can be maintained in MODE 3 from the AHSDPs and local control stations.

The 18 month Frequency is based upon the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Operating experience demonstrates that Alternate Shutdown System instrumentation channels seldom fail to pass the Surveillance when performed at a Frequency of once every 18 months. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. Note 1 states that the SR is not required for Functions 16, 17, and 18; Note 2 states that it is not necessary to calibrate neutron detectors because of the difficulty of simulating a meaningful signal. Wide range and source range nuclear instrument channels are not calibrated to indicate the actual power level or the flux in the detector location. The circuitry is adjusted so that wide range and source range readings may be used to determine the approximate reactor flux level for comparative purposes.

REFERENCES

1. FSAR, Section 7.4, "Other Safety Related Protection, Control, and Display Systems"

2. 10 CFR 50, Appendix A, GDC 19 and Appendix R.

B 3.3 INSTRUMENTATION

B 3.3.9 Neutron Flux Monitoring Channels

BASES

BACKGROUND

The neutron flux monitoring channels consist of two combined source range/wide range channels, designated NI-1/3 and NI-2/4. The wide range portions, (NI-3 and NI-4) provide neutron flux power indication from < 1E-7% RTP to > 100% RTP. The source range portions, designated NI-1 and NI-2, provide source range indication over the range of 0.1 to 1E+5 cps.

This LCO addresses MODES 3, 4, and 5. In MODES 1 and 2, the neutron flux monitoring requirements are addressed by LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation."

When the plant is shutdown, both neutron flux monitoring channels must be available to monitor neutron flux. If only one section of a neutron flux monitoring channel (source range or wide range) is functioning, the neutron flux monitoring channel may be considered OPERABLE if it is capable of detecting the existing reactor neutron flux. In this application, the RPS channels need not be OPERABLE since the reactor trip Function is not required. By monitoring neutron flux, loss of SDM caused by boron dilution can be detected as an increase in flux. Two channels must be OPERABLE to provide single failure protection and to facilitate detection of channel failure by providing CHANNEL CHECK capability.

APPLICABLE SAFETY ANALYSES

The neutron flux monitoring channels are necessary to monitor core reactivity changes. They are the primary means for detecting, and triggering operator actions to respond to, reactivity transients initiated from conditions in which the RPS is not required to be OPERABLE. The neutron flux monitoring channel's LCO requirements support compliance with 10 CFR 50, Appendix A, GDC 13 (Ref. 1). The FSAR, Chapters 7 and 14 (Refs. 2 and 3, respectively), describes the specific neutron flux monitoring channel features that are critical to comply with the GDC.

BASES

APPLICABLE SAFETY ANALYSIS (continued)	The OPERABILITY of neutron flux monitoring channels is necessary to meet the assumptions of the safety analyses and provide for the detection of reduced SDM. The neutron flux monitoring channels satisfy Criterion 4 of 10 CFR 50.36(c)(2).
LCO	The LCO on the neutron flux monitoring channels ensures that adequate information is available to verify core reactivity conditions while shut down. The safety function of these instruments is to detect changes in core reactivity such as might occur from an inadvertent boron dilution. Two neutron flux monitoring channels are required to be OPERABLE. If only one section of a neutron flux monitoring channel (source range or wide range) is functioning, the neutron flux monitoring channel may be considered OPERABLE if it is capable of detecting the existing reactor neutron flux. For example, with the source range count rate indicator functioning properly within its range, and in reasonable agreement with the other source range, a neutron flux monitor channel may be considered OPERABLE even though its wide range indicator is not functioning. The source range nuclear instrumentation channels, NI-1 and NI-2, provide neutron flux coverage extending an additional one to two decades below the wide range channels for use during refueling, when neutron flux may be extremely low. This LCO does not require OPERABILITY of the High Startup Rate Trip Function or the Zero Power Mode Bypass Removal Function. Those functions are addressed in LCO 3.3.1, RPS Instrumentation.
APPLICABILITY	In MODES 3, 4, and 5, neutron flux monitoring channels must be OPERABLE to monitor core power for reactivity changes. In MODES 1 and 2, neutron flux monitoring channels are addressed as part of the RPS in LCO 3.3.1. The requirements for source range neutron flux monitoring in MODE 6 are addressed in LCO 3.9.2, "Nuclear Instrumentation."

BASES

ACTIONS

A.1 and A.2

With one required channel inoperable, it may not be possible to perform a CHANNEL CHECK to verify that the other required channel is OPERABLE. Therefore, with one or more required channels inoperable, the neutron flux power monitoring Function cannot be reliably performed. Consequently, the Required Actions are the same for one required channel inoperable or more than one required channel inoperable. The absence of reliable neutron flux indication makes it difficult to ensure SDM is maintained. Required Action A.1, therefore, requires that all positive reactivity additions that are under operator control, such as boron dilution or PCS temperature changes, be halted immediately, preserving SDM.

SDM must be verified periodically to ensure that it is being maintained. The initial Completion Time of 4 hours and once every 12 hours thereafter to perform SDM verification takes into consideration that Required Action A.1 eliminates many of the means by which SDM can be reduced. These Completion Times are also based on operating experience in performing the Required Actions and the fact that plant conditions will change slowly.

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per. . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.3.9 A.2 must be initially performed within 4 hours without any SR 3.0.2 extension, subsequent performances at the "Once per 12 hours" interval may utilize the 25% SR 3.0.2 extension.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

SR 3.3.9.1 is the performance of a CHANNEL CHECK on each required channel every 12 hours. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based upon the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff and should be based on a combination of the channel instrument uncertainties including indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of displays associated with the LCO required channels.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.2

SR 3.3.9.2 is the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is performed every 18 months. The Surveillance is a complete check and readjustment of the neutron flux channel from the preamplifier input through to the remote indicators.

This SR is modified by a Note which states that it is not necessary to calibrate neutron detectors because of the difficulty of simulating a meaningful signal. Wide range and source range nuclear instrument channels are not calibrated to indicate the actual power level or the flux in the detector location. The circuitry is adjusted so that wide range and source range readings may be used to determine the approximate reactor flux level for comparative purposes.

This LCO does not require the OPERABILITY of the High Startup Rate trip function or the Zero Power Mode Bypass removal function. The OPERABILITY of those functions does not have to be verified during performance of this SR. Those functions are addressed in LCO 3.3.1, RPS Instrumentation.

This Frequency is the same as that employed for the same channels in the other applicable MODES.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13

2. FSAR, Chapter 7

3. FSAR, Chapter 14

B 3.3 INSTRUMENTATION

B 3.3.10 Engineered Safeguards Room Ventilation (ESRV) Instrumentation

BASES

BACKGROUND

This LCO addresses the instrumentation which provides isolation of the ESRV System (Ref. 1). The ESRV Instrumentation high radiation signal provides automatic damper closure, using two radiation monitors. One radiation monitor is located in the ventilation system duct work associated with each of the Engineered Safeguards (ES) pump rooms. Upon detection of high radiation, the ESRV Instrumentation actuates isolation of the associated ES pump room by closing the dampers in the ventilation system inlet and discharge paths. Typically, high radiation would only be expected due to excessive leakage during the recirculation phase of operation following a Loss of Coolant Accident (LOCA). The ESRV System is addressed by LCO 3.7.13, "Engineered Safeguards Room Ventilation (ESRV) Dampers."

APPLICABLE SAFETY ANALYSES

The ESRV Instrumentation isolates the ES pump rooms in the event of high radiation in the pump rooms due to leakage during the recirculation phase. The analysis for a Maximum Hypothetical Accident (MHA) described in FSAR, Section 14.22 (Ref. 2), assumes a reduction factor in the potential radioactive releases from the ES pump rooms due to plateout following automatic isolation. However, no specific value is assumed in the MHA for the actuation of the isolation. The results indicate that the potential MHA offsite doses would be less than 10 CFR 100 guidelines.

The ESRV Instrumentation satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The LCO for the ESRV Instrumentation requires both channels to be OPERABLE to initiate ES pump room isolation when high radiation exceeds the trip setpoint.

The ESRV Instrumentation Setpoint is specified as $\leq$ 2.2E+5 cpm. This setpoint is high enough to avoid inadvertent actuation in the event of normal background radiation fluctuations during testing, but low enough to isolate the ES pump room in the event of radiation levels indicative of a LOCA and excessive leakage during recirculation of primary coolant through the ES pump room.

BASES

APPLICABILITY

The ESRV Instrumentation must be OPERABLE in MODES 1, 2, 3, and 4. In these MODES, the potential exists for an accident that could release fission product radioactivity into the primary coolant which could subsequently be released to the environment by leakage from the ES systems which are recirculating the coolant.

While in MODE 5 and in MODE 6, the ESRV Instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the 10 CFR 100 guidelines.

ACTIONS

The most common causes of channel inoperability are outright failure of loop components or drift of those loop components which is sufficient to exceed the tolerance provided in the plant setpoint analysis. Loop component failures are typically identified by the actuation of alarms due to the channel failing to the "safe" condition, during CHANNEL CHECKS (when the instrument is compared to the redundant channels), or during the CHANNEL FUNCTIONAL TEST (when an automatic component might not respond properly). Typically, the drift of the loop components is found to be small and results in a delay of actuation rather than a total loss of function. Excessive loop component drift would, most likely, be identified during a CHANNEL CHECK (when the instrument is compared to the redundant channels) or during a CHANNEL CALIBRATION (when instrument loop components are checked against reference standards).

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each channel since each channel serves to isolate a different Engineered Safeguards Room. The Completion Times of each inoperable channel will be tracked separately, starting from the time the Condition was entered.

<u>A.1</u>

Condition A addresses the failure of one or both ESRV Instrumentation high radiation monitoring channels. Operation may continue as long as action is immediately initiated to isolate the ESRV System. With the inlet and exhaust dampers closed, the ESRV Instrumentation is no longer required since the potential pathway for radioactivity to escape to the environment has been removed.

BASES

ACTIONS

A.1 (continued)

The Completion Time for this Required Action is commensurate with the importance of maintaining the ES pump room atmosphere isolated from the outside environment when the ES pumps are circulating primary coolant.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.10.2

A CHANNEL FUNCTIONAL TEST is performed on each ESRV Instrumentation channel to ensure the entire channel will perform its intended function. 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 Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Any setpoint adjustment must be consistent with the assumptions of the setpoint analyses.

The Frequency of 31 days is based on plant operating experience with regard to channel OPERABILITY, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is a rare event.

SR 3.3.10.3

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the setpoint analysis.

The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. FSAR, Section 7.4.5.2
2. FSAR, Section 14.22

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

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

BASES

BACKGROUND

These Bases address requirements for maintaining PCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters, when appropriate measurement uncertainties are applied, 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.

Another set of limits on DNB related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core Safety Limits." The restriction of the SLs prevent overheating of the fuel and cladding that would result in the release of fission products to the primary coolant. The limits of LCO 3.4.1, in combination with other LCOs, are designed to prevent violation of the reactor core SLs.

The LCO limits for minimum and maximum PCS pressures as measured at the pressurizer are consistent with operation within the nominal operating envelope and are bounded by those used as the initial pressures in the analyses.

The LCO limit for maximum PCS cold leg temperature is consistent with operation at steady state power levels and is bounded by those used as the initial temperatures in the analyses.

The LCO limits for minimum PCS flow rate is bounded by those used as the initial flow rates in the analyses. The PCS flow rate is not expected to vary during plant operation with all Primary Coolant Pumps running.

BASES

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB DNB limited transients analyzed in the safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR Safety Limit (SL 2.1.1). This is the acceptance limit for the PCS DNB parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed for include loss of coolant flow events and dropped or stuck control rod 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, "Regulating Rod Group Position Limits"; LCO 3.2.3, "Quadrant Power Tilt"; and LCO 3.2.4, "AXIAL SHAPE INDEX." The safety analyses are performed over the following range of initial values: PCS pressure 1700 - 2300 psia, core inlet temperature 500-580°F, and a measured reactor vessel inlet coolant flow rate ≥ 352,000 gpm.

The PCS DNB limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

LCO

This LCO specifies limits on the monitored process of variables PCS pressurizer pressure and PCS cold leg temperature, and the calculated value of PCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. 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 are given for the measurement location but have not been adjusted for instrument error. Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of this LCO. Instrument errors and the PCS flow rate measurement error are applied to the LCO numerical values in the safety analysis.

BASES

LCO (continued)

LCO 3.4.1.b is modified by a Note which states if the measured primary coolant system flow rate is greater the 150.0 E6 lbm/hr, then the maximum cold leg temperature shall be less than or equal to the T_c derived at 150.0 E6 lbm/hr. The purpose of this Note is to restrict the calculated value of T_c to within the validity limits of the T_c equation. A DNB analysis was performed in a parametric fashion to determine the core inlet temperature as a function of pressure and flow which the minimum DNBR is equal to the DNB correlation safety limit. This analysis includes the following uncertainties and allowances: 2% of rated power for power measurement; ± 0.06 for ASI measurement; ± 22 psi for pressurizer pressure; ± 2°F for inlet temperature; and 3% measurement and 3% bypass for core flow. In addition, transient biases were included in the determination of the allowable reactor inlet temperature. The limits of validity of the T_c equation are:

Pressurizer Pressure ≥ 1800 and ≤ 2200 psia;
PCS Flow Rate ≥ 100.0 E6 and ≤ 150.0 E6 lbm/hr; and
ASI as shown in COLR.

Thus, limiting the maximum allowed T_c to the value derived at 150.0 E6 lbm/hr assures an increase in the margin to DNB for PCS flow rates in excess of 150.0 E6 lbm/hr.

APPLICABILITY

In MODE 1, the limits on PCS pressurizer pressure, PCS cold leg temperature, and PCS 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.

ACTIONS

A.1

Pressurizer pressure and cold leg temperature are controllable and measurable parameters. PCS flow rate is not a controllable parameter and is not expected to vary during steady state operation. With any of these parameters not within the LCO limits, action must be taken to restore the parameter.

The 2 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.

BASES

ACTIONS (continued)

B.1

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

Six hours is a reasonable time that permits the plant power to be reduced at an orderly rate without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 and SR 3.4.1.2

The Surveillance for monitoring pressurizer pressure and PCS cold leg temperature is performed using installed instrumentation. 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 3.4.1.3

Measurement of PCS total flow rate verifies that the actual PCS flow rate is within the bounds of the analyses. This verification may be performed by a calorimetric heat balance or other method.

The Frequency of 18 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. PCS flow rate must also be verified after plugging of each 10 or more steam generator tubes since plugging 10 or more tubes could result in an increase in PCS flow resistance. Plugging less than 10 steam generator tubes will not have a significant impact on PCS flow resistance and, as such, does not require a verification of PCS flow rate.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.3 (continued)

The SR is modified by a Note that states the SR is only required to be performed 31 EFPD after THERMAL POWER is ≥ 90% RTP. The Note is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1. The most common, and perhaps accurate, method used to perform the PCS total flow surveillance is by means of a primary to secondary heat balance (calorimetric) with the plant at or near full rated power. The most accurate results for such a test are obtained with the plant at or near full power when differential temperatures measured across the reactor are the greatest. Consequently, the test should not be performed until reaching near full power (i.e., ≥ 90% RTP) conditions. Similarly, test accuracy is also influenced by plant stability. In order for accurate results to be obtained, steady state plant conditions must exist to permit meaningful data to be gathered during the test. Typically, following an extended shutdown the secondary side of the plant will take up to several days to stabilize after power escalation. It is impracticable to perform a primary to secondary heat balance of the precision required for the PCS flow measurement until stabilization has been achieved. Furthermore, an integral part of the PCS flow heat balance involves the use of Ultrasonic Flow Measurement equipment for measuring steam generator feedwater flow. This equipment requires, stable plant operation at or near full power conditions before it can be used. As such, the Surveillance cannot be performed in MODE 2 or below, and will not yield accurate results if performed below 90% RTP.

REFERENCES

1. FSAR, Section 14.1

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.2 PCS 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 primary coolant moderator temperature coefficient used in core operating and accident analysis is typically defined for the normal operating temperature range (532°F to 570°F). The Reactor Protective System receives inputs from the narrow range hot leg and cold leg temperature instruments, which have a range of 515°F to 615°F. The PCS loop average temperature (T_{ave}) is controlled using inputs of the same range. Nominal T_{ave} for making the reactor critical is 532°F. Safety and operating analyses for lower than 525°F have not been made.

APPLICABLE SAFETY ANALYSES

There are no accident analyses that dictate the minimum temperature for criticality, but existing transient analysis are bounding for operation at low power with cold leg temperatures of 525°F (Ref. 1).

The PCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The purpose of the LCO is to prevent criticality outside the normal operating regime (532°F to 570°F) and to prevent operation in an unanalyzed condition.

The LCO provides a reasonable distance between the hot zero power value of 532°F and the limit of 525°F. This allows adequate time to trend its approach and take corrective actions prior to exceeding the limit.

BASES

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 $K_{eff} \geq 1.0$.
ACTIONS	<u>A.1</u> If T_{ave} is below 525°F and cannot be restored in 30 minutes, 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	<u>SR 3.4.2.1</u> PCS loop average temperature is required to be verified at or above 525°F every 12 hours. The SR to verify PCS loop average temperature every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room.
REFERENCES	1. FSAR, Section 14.1.3

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

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

BASES

BACKGROUND

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

Figures 3.4.3-1 and 3.4.3-2 contain P/T limit curves for heatup, cooldown, and Inservice Leak and Hydrostatic (ISLH) testing, and data for the maximum rate of change of primary coolant temperature. A discussion of the methodology for the development of the P/T curves is provided in Reference 1.

Each P/T limit curve defines an acceptable region for normal operation. The P/T limit curves include an allowance to account for the fact that pressure is measured in the pressurizer rather than at the vessel beltline and to account for primary coolant pump discharge pressure. The use of the curves provides operational limits during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the Primary Coolant Pressure Boundary (PCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply to the vessel.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the PCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 3).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

BASES

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

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

The minimum temperature at which the reactor can be made critical, as required by Reference 2, shall be at least 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, "PCS Minimum Temperature for Criticality," and LCO 3.1.7, "Special Test Exceptions (STE)."

The consequence of violating the LCO limits is that the PCS has been operated under conditions that can result in brittle failure of the PCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the PCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), 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 PCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. 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 themselves since they preclude operation in an unanalyzed condition.

The PCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

BASES

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 PCS, 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.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other PCPB 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 PCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

BASES

APPLICABILITY (continued)

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, "PCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "PCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature 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 PCPB 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 PCS operation can continue. The evaluation must verify the PCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), 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.

BASES

ACTIONS

A.1 and A.2 (continued)

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 PCPB 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 PCS 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 PCS at reduced pressure and temperature. With reduced pressure and temperature 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 PCS pressure < 270 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.

BASES

ACTIONS (continued)

C.1 and C.2

The actions of this LCO, anytime other than in MODE 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 PCPB 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 PCS operation can continue. The evaluation must verify that the PCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), 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 PCPB integrity.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the limits of Figure 3.4.3-1 and Figure 3.4.3-2 is required every 30 minutes when PCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor PCS 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. Calculation of the average hourly cooldown rate must consider changes in reactor vessel inlet temperature caused by initiating shutdown cooling, by starting primary coolant pumps with a temperature difference between the steam generator and PCS, or by stopping primary coolant pumps with shutdown cooling in service.

Surveillance for heatup and cooldown operations may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

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

REFERENCES

1. Safety Evaluation for Palisades Nuclear Plant License Amendment No. 163, dated March 2, 1995
2. 10 CFR 50, Appendix G
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G
4. ASTM E 185-82, July 1982
5. 10 CFR 50, Appendix H
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.4 PCS Loops - MODES 1 and 2

BASES

BACKGROUND

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

The secondary functions of the PCS 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 plant shutdown.

The PCS configuration for heat transport uses two PCS loops. Each PCS loop contains an SG and two Primary Coolant Pumps (PCPs). A PCP 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 Departure from Nucleate Boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two PCS loops with both PCPs in operation in each loop. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying two PCS loops provides the minimum necessary paths (two SGs) for heat removal.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the Design Bases Accident (DBA) initial conditions including PCS pressure, PCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the primary coolant forced flow rate, which is represented by the number of PCS loops in service.

BASES

APPLICABLE SAFETY ANALYSES (continued)	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 PCPs are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to PCP operation are the Loss of Forced Primary Coolant Flow, Primary Coolant Pump Rotor Seizure and Uncontrolled Control Rod Withdrawal events (Ref. 1). Steady state DNB analysis had been performed for the four pump combination. The steady state DNB analysis, which generates the pressure and temperature and Safety Limit (i.e., the Departure from Nucleate Boiling Ratio (DNBR) limit), assumes a maximum power level of 112% RTP. This is the design overpower condition for four pump operation. The 112% value is the accident analysis setpoint of the trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit. PCS Loops - MODES 1 and 2 satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2).
LCO	The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both PCS loops with both PCPs 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 PCPs providing forced flow for heat transport to an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. SG, and hence PCS loop OPERABILITY with regards to SG water level is ensured by the Reactor Protection System (RPS) in MODES 1 and 2. A reactor trip places the plant in MODE 3 if any SG water level is ≤ 25.9% (narrow range) as sensed by the RPS. The minimum level to declare the SG OPERABLE is 25.9% (narrow range). In MODES 1 and 2, the reactor can be critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all PCS loops are required to be in operation in these MODES to prevent DNB and core damage.

BASES

APPLICABILITY

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, "PCS Loops-MODE 3";

LCO 3.4.6, "PCS Loops-MODE 4";

LCO 3.4.7, "PCS Loops-MODE 5, Loops Filled";

LCO 3.4.8, "PCS Loops-MODE 5, Loops Not Filled";

LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation-High Water Level" (MODE 6); and

LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits. It should be noted that the reactor will trip and place the plant in MODE 3 as soon as the RPS senses less than four PCPs operating.

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours of the required number of loops in operation. Verification may include indication of PCS flow, temperature, or pump status, 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. FSAR, Section 14.1

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.5 PCS Loops - MODE 3

BASES

BACKGROUND

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

In MODE 3, Primary Coolant Pumps (PCPs) 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 PCS loop with one PCP is sufficient to remove core decay heat. However, two PCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Any combination of OPERABLE PCPs and OPERABLE PCS loops can be used to fulfill the heat removal function.

Primary 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 PCS cannot be ensured. Any combination of OPERABLE PCPs and OPERABLE PCS loops can be used to fulfill the mixing function.

APPLICABLE SAFETY ANALYSES

Failure to provide heat removal may result in challenges to a fission product barrier. The PCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

PCS Loops - MODE 3 satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The purpose of this LCO is to require two PCS 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 (> -84% water level) of transferring heat from the primary coolant at a controlled rate. Forced primary coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running PCP meets the LCO requirement for one loop in operation.

BASES

LCO (continued)

Note 1 permits all PCPs to not be in operation ≤ 1 hour per 8 hour period. This means that natural circulation has been established using the SGs. The Note prohibits boron dilution 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 PCS without the PCPs depends on the core decay heat load and the length of time that the pumps are stopped. As decay heat diminishes, the effects on PCS 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 (Pressure and Temperature (P/T) limits or Low Temperature Overpressure Protection (LTOP) limits) must be observed and forced flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping the PCPs 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.

In MODE 3, it is sometimes necessary to stop all PCP forced circulation. This is permitted to perform surveillance or startup testing, to perform the transition to and from SDC, or to avoid operation below the PCP minimum net positive suction head limit. The time period is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

Note 2 requires that one of the following conditions be satisfied before forced circulation (starting the first PCP) may be started:

a. PCS cold leg temperature (T_c) is > 430°F;

b. SG secondary temperature is equal to or less than the reactor inlet temperature (T_c);

c. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and PCS heatup/cooldown rate is ≤ 10°F/hour; or

BASES

LCO (continued)

d. SG secondary temperature is < 100 °F above T_c, and shutdown cooling is isolated from the PCS, and pressurizer level is ≤ 57%.

Satisfying any of the above conditions will preclude a large pressure surge in the PCS when the PCP is started. Energy additions from the steam generators could occur if a PCP was started when the steam generator secondary temperature is significantly above the PCS temperature. The maximum pressurizer level at which credit is taken for having a bubble (57%, which provides about 700 cubic feet of steam space) is based on engineering judgement and verified by LTOP analysis. This level provides the same steam volume to dampen pressure transients as would be available at full power.

An OPERABLE PCS loop consists of any one (of the four) OPERABLE PCP and an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program and has the minimum water level specified in SR 3.4.5.2. A PCP 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 PCS loop in operation is adequate for transport and heat removal. A second PCS loop is required to be OPERABLE but is not required to be in operation for redundant heat removal capability.

Operation in other MODES is covered by:

LCO 3.4.4, "PCS Loops-MODES 1 and 2";

LCO 3.4.6, "PCS Loops-MODE 4";

LCO 3.4.7, "PCS Loops-MODE 5, Loops Filled";

LCO 3.4.8, "PCS Loops-MODE 5, Loops Not Filled";

LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation-High Water Level" (MODE 6); and

LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation-Low Water Level" (MODE 6).

BASES

ACTIONS

A.1

If one required PCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the required PCS loop to OPERABLE status within a Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, non-operating 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. Mechanical system LCOs typically provide a 72 hour Completion Time under conditions when a required system can perform its required safety function, but may not be able to do so assuming an additional failure. When operating in accordance with the Required Actions of an LCO Condition, it is not necessary to be able to cope with an additional single failure.

B.1

If restoration is not possible within 72 hours, the plant must be placed in MODE 4 within 24 hours. In MODE 4, the plant may be placed on the SDC System. The Completion Time of 24 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 PCS loop is in operation, except as provided in Note 1 in the LCO section, all operations involving a reduction of PCS boron concentration must be immediately suspended. This is necessary because boron dilution requires forced circulation for proper homogenization. Action to restore one PCS loop to OPERABLE status and operation shall be initiated immediately and continued until one PCS loop is restored to OPERABLE status and operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required number of PCS loops are in operation. Verification include indication of PCS flow, temperature, and pump status, which help ensure that forced flow is providing heat removal and mixing of the soluble boric acid. 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.

SR 3.4.5.2

This SR requires verification every 12 hours that the secondary side water level in each SG is ≥ -84% using the wide range level instrumentation. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the primary 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 PCP is OPERABLE ensures that the single failure criterion is met and that an additional PCS loop can be placed in operation, if needed, to maintain decay heat removal and primary coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required PCP that is not in operation such that the PCP is capable of being started and providing forced PCS flow if needed. Proper breaker alignment and power availability means the breaker for the required PCP is racked-in and electrical power is available to energize the PCP motor. The Frequency of 7 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 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.6 PCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the primary coolant is the removal of decay heat and transfer of this heat to the Steam Generators (SGs) or Shutdown Cooling (SDC) heat exchangers. The secondary function of the primary coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either Primary Coolant Pumps (PCPs) or SDC trains can be used for coolant circulation. The intent of this LCO is to provide forced flow from any one (of the four) PCP or one SDC train for decay heat removal and transport. The flow provided by one PCP loop or SDC train 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.

APPLICABLE SAFETY ANALYSES

The boron concentration must be uniform throughout the PCS volume to prevent stratification of primary coolant at lower boron concentrations which could result in a reactivity insertion. Sufficient mixing of the primary coolant is assured if one PCP is in operation. PCS circulation is considered in the determination of the time available for mitigation of the inadvertent boron dilution event. By imposing a minimum flow through the reactor core of 2810 gpm, sufficient time is provided for the operator to terminate a boron dilution under asymmetric flow conditions. Due to its system configuration (i.e., no throttle valves) and large volumetric flow rate, a minimum flow rate is not imposed on the PCPs.

PCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2).

LCO

The purpose of this LCO is to require that two loops or trains, PCS or SDC, be OPERABLE in MODE 4 and one of these loops or trains to be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of PCS and SDC System loops. Any one PCS loop in operation, or SDC in operation with a flow $\geq$ 2810 gpm through the reactor core, provides enough flow to remove the decay heat from the core with forced circulation and provide sufficient mixing of the soluble boric acid. An additional loop or train is required to be OPERABLE to provide redundancy for heat removal.

BASES

LCO

A SDC train may be considered OPERABLE (but not necessarily in operation) during re-alignment to, and when it is re-aligned for, LPSI service or for testing, if it is capable of being (locally or remotely) realigned to the SDC mode of operation and is not otherwise inoperable. Since SDC is a manually initiated system, it need not be considered inoperable solely because some additional manual valve realignments must be made in addition to the normal initiation actions. Because of the dual functions of the components that comprise the LPSI and shutdown cooling systems, the LPSI alignment may be preferred.

Note 1 permits all PCPs and SDC pumps to not be in operation ≤ 1 hour per 8 hour period. The Note prohibits boron dilution 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 PCS without the PCPs 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 PCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the primary coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (Pressure and Temperature (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 PCPs 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.

In MODE 4, it is sometimes necessary to stop all PCPs or SDC forced circulation. This is permitted to change operation from one SDC train to the other, perform surveillance or startup testing, perform the transition to and from SDC, or to avoid operation below the PCP minimum net positive suction head limit. The time period is acceptable because natural circulation is acceptable for decay heat removal, the primary coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

BASES

LCO (continued)

Note 2 requires that one of the following conditions be satisfied before forced circulation (starting the first PCP) may be started:

a. SG secondary temperature is $\leq T_c$;

b. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and PCS heatup/cooldown rate is ≤ 10°F/hour; or

c. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and pressurizer level is ≤ 57%.

Satisfying any of the above conditions will preclude a large pressure surge in the PCS when the PCP is started. Energy additions from the steam generators could occur if a PCP was started when the steam generator secondary temperature is significantly above the PCS temperature. The maximum pressurizer level at which credit is taken for having a bubble (57%, which provides about 700 cubic feet of steam space) is based on engineering judgement and verified by LTOP analysis. This level provides the same steam volume to dampen pressure transients as would be available at full power.

Note 3 specifies a limitation on the simultaneous operation of primary coolant pumps P-50A and P-50B which allows the pressure limits in LCO 3.4.3, "PCS Pressure and Temperature Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection System," to be higher than they would be without this limit. This is because the pressure in the reactor vessel downcomer region when primary coolant pumps P-50A and P-50B are operated simultaneously is higher than the pressure for other two primary coolant pump combinations.

An OPERABLE PCS loop consists of any one (of the four) OPERABLE PCP and an SG that has the minimum water level specified in SR 3.4.6.2 and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. PCPs are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. SDC pumps are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

BASES

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 PCS loops and SGs, or the SDC System.

Operation in other MODES is covered by:

LCO 3.4.4, "PCS Loops-MODES 1 and 2";

LCO 3.4.5, "PCS Loops-MODE 3";

LCO 3.4.7, "PCS Loops-MODE 5, Loops Filled";

LCO 3.4.8, "PCS Loops-MODE 5, Loops Not Filled";

LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation-High Water Level" (MODE 6); and

LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

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

B.1

If only one SDC train is OPERABLE and in operation with no OPERABLE PCS loops, 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 train OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining SDC train, it would be safer to initiate that loss from MODE 5 (≤ 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 train operating, in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

C.1, C.2.1, and C.2.2

If no PCS loops or SDC trains are OPERABLE, or no PCS loop is in operation and the SDC flow through the reactor core is < 2810 gpm, except during conditions permitted by Note 1 in the LCO section, all operations involving reduction of PCS boron concentration must be suspended. Action to restore one PCS loop or SDC train to OPERABLE status and operation shall be initiated immediately and continue until one loop or train is restored to operation and flow through the reactor core is restored to ≥ 2810 gpm. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one required loop or train is in operation. This ensures forced flow is providing heat removal and mixing of the soluble boric acid. Verification may include flow rate (SDC only), or indication of flow, temperature, or pump status for the PCP. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess PCS loop/SDC train status. In addition, control room indication and alarms will normally indicate loop/train status.

SR 3.4.6.2

This SR requires verification every 12 hours of secondary side water level in the required SG(s) ≥ -84% using the wide range level instrumentation. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the primary coolant. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify SG status.

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.4.6.3 Verification that the required pump is OPERABLE ensures that an additional PCS loop or SDC train can be placed in operation, if needed to maintain decay heat removal and primary coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump that is not in operation such that the pump is capable of being started and providing forced PCS flow if needed. Proper breaker alignment and power availability means the breaker for the required pump is racked-in and electrical power is available to energize the pump motor. The Frequency of 7 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 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.7 PCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the PCS loops filled, the primary function of the primary coolant is the removal of decay heat and transfer this heat either to the Steam Generator (SG) secondary side coolant via natural circulation (Ref. 1) or the Shutdown Cooling (SDC) heat exchangers. While the principal means for decay heat removal is via the SDC System, the SGs via natural circulation are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary side water. If heatup of the PCS were to continue, the contained inventory of the SGs would be available to remove decay heat by producing steam. As long as the SG secondary side water is at a lower temperature than the primary coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the primary coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with PCS loops filled, the SDC trains are the principal means for decay heat removal. The number of trains in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC train for decay heat removal and transport. The flow provided by one SDC train 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 train that must be OPERABLE and in operation. The second path can be another OPERABLE SDC train, or through the SGs, via natural circulation each having an adequate water level. "Loops filled" means the PCS loops are intact, not blocked by dams, and totally filled with coolant.

BASES

APPLICABLE SAFETY ANALYSES

The boron concentration must be uniform throughout the PCS volume to prevent stratification of primary coolant at lower boron concentrations which could result in a reactivity insertion. Sufficient mixing of the primary coolant is assured if one SDC pump is in operation. PCS circulation is considered in the determination of the time available for mitigation of the inadvertent boron dilution event. By imposing a minimum flow through the reactor core of 2810 gpm, sufficient time is provided for the operator to terminate a boron dilution under asymmetric flow conditions.

PCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2).

LCO

The purpose of this LCO is to require one SDC train be OPERABLE and in operation with either an additional SDC train OPERABLE or the secondary side water level of each SG ≥ -84%. SDC in operation with a flow through the reactor core ≥ 2810 gpm, provides enough flow to remove the decay heat from the core with forced circulation and provide sufficient mixing of the soluble boric acid. The second SDC train is normally maintained OPERABLE as a backup to the operating SDC train to provide redundant paths for decay heat removal. However, if the standby SDC train is not OPERABLE, a sufficient alternate method to provide redundant paths for decay heat removal is two SGs with their secondary side water levels ≥ -84%. Should the operating SDC train fail, the SGs could be used to remove the decay heat via natural circulation.

A SDC train may be considered OPERABLE (but not necessarily in operation) during re-alignment to, and when it is re-aligned for, LPSI service or for testing, if it is capable of being (locally or remotely) realigned to the SDC mode of operation and is not otherwise inoperable. Since SDC is a manually initiated system, it need not be considered inoperable solely because some additional manual valve realignments must be made in addition to the normal initiation actions. Because of the dual functions of the components that comprise the LPSI and shutdown cooling systems, the LPSI alignment may be preferred.

Note 1 permits all SDC pumps to not be in operation ≤ 1 hour per 8 hour period. The Note prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least

BASES

LCO (continued)

10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the PCS without the 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 PCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the primary coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (Pressure and Temperature (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.

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

Note 2 allows both SDC trains to be inoperable for a period of up to 2 hours provided that one SDC train is in operation providing the required flow, the core outlet temperature is at least 10°F below the corresponding saturation temperature, and each SG secondary water level is ≥ 84%. This permits periodic surveillance tests or maintenance to be performed on the inoperable trains during the only time when such evolutions are safe and possible.

Note 3 requires that one of the following conditions be satisfied before forced circulation (starting the first PCP) may be started:

a. SG secondary temperature is equal to or less than the reactor inlet temperature (T_c);

b. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and PCS heatup/cooldown rate is ≤ 10°F/hour; or

c. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and pressurizer level is ≤ 57%.

BASES

LCO
(continued)

Satisfying any of the above conditions will preclude a large pressure surge in the PCS when the PCP is started. Energy additions from the steam generators could occur if a PCP was started when the steam generator secondary temperature is significantly above the PCS temperature. The maximum pressurizer level at which credit is taken for having a bubble (57%, which provides about 700 cubic feet of steam space) is based on engineering judgement and verified by LTOP analysis. This level provides the same steam volume to dampen pressure transients as would be available at full power.

Note 4 specifies a limitation on the simultaneous operation of primary coolant pumps P-50A and P-50B which allows the pressure limits in LCO 3.4.3, "PCS Pressure and Temperature Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection System," to be higher than they would be without this limit.

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

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. SDC pumps are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

An SG can perform as a heat sink via natural circulation when it has the minimum water level specified in SR 3.4.7.2 and is OPERABLE in accordance with the SG Tube Surveillance Program.

BASES

APPLICABILITY

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

Operation in other MODES is covered by:

LCO 3.4.4, "PCS Loops-MODES 1 and 2";

LCO 3.4.5, "PCS Loops-MODE 3";

LCO 3.4.6, "PCS Loops-MODE 4";

LCO 3.4.8, "PCS Loops-MODE 5, Loops Not Filled";

LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation-High Water Level" (MODE 6); and

LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation-Low Water Level" (MODE 6).

BASES

ACTIONS

A.1 and A.2

If one SDC train is inoperable and any SG has a secondary side water level < -84%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second SDC train 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 trains are OPERABLE or SDC flow through the reactor core is < 2810 gpm, except as permitted in Note 1, all operations involving the reduction of PCS boron concentration must be suspended. Action to restore one SDC train to OPERABLE status and operation shall be initiated immediately and continue until one train is restored to operation and flow through the reactor core is restored to ≥ 2810 gpm. Boron dilution requires forced circulation for proper mixing and the margin to criticality must not be reduced in this type of operation. 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 train is in operation. Verification of the required flow rate ensures forced flow is providing heat removal and mixing of the soluble boric acid. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess SDC train status. In addition, control room indication and alarms will normally indicate train status.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.2

This SR requires verification every 12 hours of secondary side water level in the required SGs ≥ -84% using the wide range level instrumentation. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the primary coolant. The Surveillance is required to be performed when the LCO requirement is being met by use of the SGs. If both SDC trains 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 SG status.

SR 3.4.7.3

Verification that the second SDC train is OPERABLE ensures that redundant paths for decay heat removal are available. The requirement also ensures that the additional train can be placed in operation, if needed, to maintain decay heat removal and primary coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump that is not in operation such that the SDC pump is capable of being started and providing forced PCS flow if needed. Proper breaker alignment and power availability means the breaker for the required SDC pump is racked-in and electrical power is available to energize the SDC pump motor. The Surveillance is required to be performed when the LCO requirement is being met by one of two SDC trains, e.g., both SGs have < -84% water level. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

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

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.8 PCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the PCS loops not filled, the primary function of the primary coolant is the removal of decay heat and transfer of this heat to the Shutdown Cooling (SDC) heat exchangers. The Steam Generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the primary coolant is to act as a carrier for the soluble neutron poison, boric acid. A loop is considered "not filled" if it has been drained so air has entered the loop which has not yet been removed.

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

APPLICABLE SAFETY ANALYSES

The boron concentration must be uniform throughout the PCS volume to prevent stratification of primary coolant at lower boron concentrations which could result in a reactivity insertion. Sufficient mixing of the primary coolant is assured if one SDC pump is in operation. PCS circulation is considered in the determination of the time available for mitigation of the inadvertent boron dilution event. By imposing a minimum flow through the reactor core of ≥ 2810 gpm, or a minimum flow through the reactor core ≥ 650 gpm with two of the three charging pumps incapable of reducing the boron concentration in the PCS below the minimum value necessary to maintain the required SHUTDOWN MARGIN, sufficient time is provided for the operator to terminate a boron dilution under asymmetric flow conditions.

PCS loops - MODE 5 (Loops Not Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2).

BASES

LCO

The purpose of this LCO is to require a minimum of two SDC trains be OPERABLE and one of these trains be in operation. SDC in operation with a flow rate through the reactor core of ≥ 2810 gpm, or with a flow rate through the reactor core of ≥ 650 gpm with two of the three charging pumps incapable of reducing the boron concentration in the PCS below the minimum value necessary to maintain the required SHUTDOWN MARGIN, provides enough flow to remove the decay heat from the core with forced circulation and provide sufficient mixing of the soluble boric acid. The restriction on charging pump operations only applies to those cases where the potential exists to reduce the PCS boron concentration below minimum the boron concentration necessary to maintain the required SHUTDOWN MARGIN. It is not the intent of this LCO to restrict charging pump operations when the source of water to the pump suction is greater than or equal to the minimum boron concentration necessary to maintain the required SHUTDOWN MARGIN. An additional SDC train is required to be OPERABLE to meet the single failure criterion.

A SDC train may be considered OPERABLE (but not necessarily in operation) during re-alignment to, and when it is re-aligned for, LPSI service or for testing, if it is capable of being (locally or remotely) realigned to the SDC mode of operation and is not otherwise inoperable. Since SDC is a manually initiated system, it need not be considered inoperable solely because some additional manual valve realignments must be made in addition to the normal initiation actions. Because of the dual functions of the components that comprise the LPSI and shutdown cooling systems, the LPSI alignment may be preferred.

Note 1 permits all SDC pumps to not be in operation for ≤ 1 hour. The Note prohibits boron dilution 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 flow obstruction. Operations which could drain the PCS and thereby cause a loss of, or failure to regain SDC capability are also prohibited.

In MODE 5 with loops not filled, it is sometimes necessary to stop all SDC forced circulation. This is permitted to change operation from one SDC train to the other, and to perform surveillance or startup testing. The time period is acceptable because the primary coolant will be maintained subcooled, and boron stratification affecting reactivity control is not expected.

BASES

LCO (continued)

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

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. SDC pumps are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

APPLICABILITY

In MODE 5 with PCS loops not filled, this LCO requires forced circulation to remove decay heat from the core and to provide proper boron mixing. One SDC train provides sufficient circulation for these purposes.

Operation in other MODES is covered by:

LCO 3.4.4, "PCS Loops-MODES 1 and 2";

LCO 3.4.5, "PCS Loops-MODE 3";

LCO 3.4.6, "PCS Loops-MODE 4";

LCO 3.4.7, "PCS Loops-MODE 5, Loops Filled";

LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation-High Water Level" (MODE 6); and

LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

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

BASES

ACTIONS (continued)

B.1 and B.2

If no SDC trains are OPERABLE or SDC flow through the reactor core is not within limits, except as provided in Note 1, all operations involving the reduction of PCS boron concentration must be suspended. Action to restore one SDC train to OPERABLE status and operation shall be initiated immediately and continue until one train is restored to operation and flow through the reactor core is restored to within limits. Boron dilution requires forced circulation for proper mixing and the margin to criticality must not be reduced in this type of operation. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1 and SR 3.4.8.2

These SRs require verification every 12 hours that one SDC train is in operation. Verification of the required flow rate ensures forced circulation is providing heat removal and mixing of the soluble boric acid. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess SDC train status. In addition, control room indications and alarms will normally indicate train status.

SR 3.4.8.1 and SR 3.4.8.2 are each modified by a Note to indicate the SR is only required to be met when complying with the applicable portion of the LCO. Therefore, it is only necessary to perform either SR 3.4.8.1, or SR 3.4.8.2 based on the method of compliance with the LCO.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.8.3

This SR requires verification every 12 hours that two of the three charging pumps are incapable of reducing the boron concentration in the PCS below the minimum value necessary to maintain the required SHUTDOWN MARGIN. Making the charging pumps incapable reducing the boron concentration in the PCS may be accomplished by electrically disabling the pump motors, blocking potential dilution sources to the pump suction, or by isolating the pumps discharge flow path to the PCS. Verification may include visual inspection of the pumps configuration (e.g., pump breaker position or valve alignment), or the use of other administrative controls. The 12 hour Frequency is based on engineering judgement considering operating practice, administrative control available, and the unlikeness of inadvertently aligning a charging pump for PCS injection during this period.

SR 3.4.8.3 is modified by a Note to indicate the SR is only required to be met when complying with LCO 3.4.8.b. When SDC flow through the reactor core is $\geq$ 2810 gpm, there is no restriction on charging pump operation.

SR 3.4.8.4

Verification that the required number of trains are OPERABLE ensures that redundant paths for heat removal are available and that additional trains can be placed in operation, if needed, to maintain decay heat removal and primary coolant circulation. Verification is performed by verifying proper breaker alignment and indicated power available to the required pump that is not in operation such that the SDC pump is capable of being started and providing forced PCS flow if needed. Proper breaker alignment and power availability means the breaker for the required SDC pump is racked-in and electrical power is available to energize the SDC pump motor. The Frequency of 7 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 PRIMARY COOLANT SYSTEMS (PCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

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

The pressure control components addressed by this LCO include the pressurizer water level, required heaters capacity, and the emergency power supply to the heaters powered from electrical bus 1E. Pressurizer safety valves and pressurizer Power Operated Relief Valves (PORVs) are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit PCS 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 primary 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 PCS pressure might exceed the Safety Limit of 2750 psia.

BASES

BACKGROUND (continued)	The requirement to have pressurizer heaters ensures that PCS pressure can be maintained. The pressurizer heaters maintain PCS pressure to keep the primary coolant subcooled. Inability to control PCS 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 FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the PCS 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 NUREG-0737, "Clarification of TMI Action Plan Requirements," is the reason for their inclusion. The intent is to keep the primary coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power. While a 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 Criterion 2 (for pressurizer water level) and Criterion 4 (for pressurizer heaters) of 10 CFR 50.36(c)(2).

BASES

LCO

The LCO requirement for the pressurizer to be OPERABLE with water level < 62.8% (hot full power pressurizer high level alarm setpoint) 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. During a plant heatup, the PCS is generally water solid in the lower temperature range of MODE 3. Therefore, LCO 3.4.9.a has been modified by a Note which states that the pressurizer water level limit does not apply in MODE 3 until after a bubble has been established in the pressurizer and the pressurizer water level has been lowered to its normal operating band. The intent of this Note is to allow entry into the mode of Applicability during a plant heatup when the pressurizer water level is above the limit specified in the LCO. Once the normal pressurizer water level is established, compliance with the LCO must be met without reliance on the Note.

The LCO requires ≥ 375 kW of pressurizer heater capacity available from electrical bus 1D, and ≥ 375 kW of pressurizer heater capacity available from electrical bus 1E with the capability of being powered from an emergency power supply. In the event of a loss of offsite power, one half of the required heater capacity is normally connected to engineered safeguards bus 1D and can be manually controlled via a hand switch in the control room. This would provide sufficient heater capacity to establish and maintain natural circulation in a hot standby condition. To provide a redundant source of heater capacity should bus 1D become unavailable, methods and procedures have been established for manually connecting the required pressurizer heaters capacity, normally fed from electrical bus 1E, to engineered safeguards electrical bus 1C via a jumper cable. The amount of time required to make this connection (less than five hours) has been evaluated to assure that a 20°F subcooling margin, due to pressure decay, is not exceeded (Ref. 2).

The value of 375 kW is derived from the use of 30 heaters rated at approximately 12.5 kW each. The actual amount needed to maintain pressure is dependent on the ambient heat losses.

BASES

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on PCS temperature resulting in the greatest effect on pressurizer level and PCS pressure control. Thus, the Applicability has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3. The purpose is to prevent water solid PCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation. Although the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," ensures overpressure protection is provided in MODE 3 when the PCS cold leg temperature is < 430°F, the Applicability for the pressurizer is all inclusive of MODE 3 since the pressurizer heaters are required in all of MODE 3 to support plant operations. In MODES 4, 5, and 6, the pressurizer is no longer required and overpressure protection is provided by LTOP components specified in LCO 3.4.12.

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 PCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Shutdown Cooling System is in service and therefore the 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 plant must be brought to MODE 3, with the reactor trippped, within 6 hours and to MODE 4 within 30 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 pressure and temperature reduction to MODE 4 brings the plant to a MODE where the LCO is not applicable. The 30 hour time to reach the nonapplicable MODE is reasonable based on operating experience for that evolution.

BASES

ACTIONS (continued)

B.1

If < 375 kW of pressurizer heater capacity is available from either electrical bus 1D or electrical bus 1E, or the pressurizer heaters from electrical bus 1E are not capable of being powered from an emergency power supply, 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 the remaining available pressurizer heaters.

C.1 and C.2

If the required pressurizer heaters cannot be restored to an OPERABLE status 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 30 hours. The Completion Time of 6 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 30 hours is reasonable, based on operating experience, to reach MODE 4 from full power in an orderly manner and without challenging plant systems.

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 is performed by observing the indicated level. SR 3.4.9.1 is modified by a Note which states that verification of the pressurizer water level is not required to be met until 1 hour after a bubble has been established in the pressurizer and the pressurizer water level has been lowered to its normal operating band. The intent of this Note is to prevent an SR 3.0.4 conflict by delaying the performance of this SR until after the water level in the pressurizer is within its normal operating band following a plant heatup. 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 capacity of the associated pressurizer heaters are verified to be ≥ 375 kW. (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 18 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

SR 3.4.9.3

This SR only applies to the pressurizer heaters normally powered from electrical bus 1E since the pressurizer heaters powered from bus 1D are permanently connected to the engineered safeguards electrical system.

This SR confirms that the pressurizer heaters normally fed from electrical bus 1E are capable of being powered from electrical bus 1C by use of a jumper cable. It is not the intent of this SR to physically install the jumper cable, but to verify the necessary components are available for installation and to ensure the procedures and methods used to install the jumper cable are current. The Frequency of 18 months is based on engineering judgement and is considered acceptable when considering the design reliability of the equipment (the jumper cable is left permanently in place and dedicated to providing the emergency feed function only), and administrative control which govern configuration management and changes to plant procedures.

REFERENCES

1. FSAR, Chapter 14
2. FSAR, Section 4.3.7

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the three spring loaded pressurizer safety valves is to provide PCS overpressure protection. Operating in conjunction with the Reactor Protection System, three valves are used to ensure that the Safety Limit (SL) of 2750 psia is not exceeded for analyzed transients during operation in MODES 1 and 2 and portions of MODE 3. For the remainder of MODE 3, MODE 4, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and the LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the American Society of Mechanical Engineering (ASME), Boiler and Pressure Vessel Code, Section III (Ref. 1). The required lift settings are given in Table 3.4.10-1 in the accompanying technical specification. The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves, acoustic monitors, and by an increase in the quench tank temperature and level.

The lift settings listed in Table 3.4.10-1 correspond to ambient conditions of the valves at nominal operating temperature and 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 PCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit (Ref. 1) could include damage to PCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES

APPLICABLE SAFETY ANALYSES

All accident analyses in the FSAR that require safety valve actuation assume operation of one or more pressurizer safety valves to limit increasing primary coolant pressure. The overpressure protection analysis assumes that the valves open at the high range of the lift setting including the allowable tolerance. The Loss of External Electrical Load incident and Loss of Normal Feedwater Flow incident are the two safety analyses events which rely on the pressurizer safety valves to mitigate an overpressurization of the PCS. The pressurizer safety valves must also accommodate pressurizer in surges that could occur from a Loss of Forced Primary Coolant Flow incident, and a Primary Pump Rotor Seizure incident. 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 Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The three pressurizer safety valves are set to open near the PCS design pressure (2500 psia) and within the ASME specified tolerance to avoid exceeding the maximum PCS design pressure SL, to maintain accident analysis assumptions, and to comply with ASME Code requirements. The nominal lift settings values listed in Table 3.4.10-1, plus an allowable tolerance of ± 3%, establish the acceptable "as-found" pressure band for determining valve OPERABILITY. Following valve testing, an as-left tolerance of ± 1% of the lift settings is imposed by SR 3.4.10.1 to account for setpoint drift during the surveillance interval. The limit protected by this specification is the Primary Coolant Pressure Boundary (PCPB) SL of 110% of design pressure. The inoperability of any valve 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 PCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

BASES

APPLICABILITY

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

The LCO is not applicable in MODE 3 when any PCS cold leg temperatures are < 430°F and MODES 4 and 5 because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head removed.

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 PCS overpressure protection system. An inoperable safety valve coincident with an PCS overpressure event could challenge the integrity of the PCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and at least one PCS cold leg temperature reduced to below 430°F within 12 hours. The 6 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 any PCS cold leg temperature < 430°F without challenging plant systems. Below 430°F, overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 3 with any PCS cold leg temperature < 430°F reduces the PCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 1), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint tolerance is ± 3% for OPERABILITY; however, the valves are reset to within a tolerance of ± 1% during the Surveillance to allow for drift.

REFERENCES

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

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

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 safety valves are addressed by LCO 3.4.10. The PORVs are solenoid-pilot operated relief valves which, when placed in the "Auto" position, automatically open at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. The PORV may also be manually operated using controls installed in the control room.

A motor operated, normally closed, block valve is installed between the pressurizer and each 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 Loss Of Coolant Accident (LOCA). Closure terminates the PCS depressurization and coolant inventory loss.

The PORV, its block valve, and their respective controls are powered from safety class power supplies. Power supplies for the PORV are separate from those for the block valve. Power supply requirements are defined in NUREG-0737, Item II.G.1.

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

In the event of an abnormal transient, the PORVs may be manually operated to depressurize the PCS as directed by the Emergency Operating Procedures. The PORVs may be used for depressurization when the pressurizer spray is not available, a condition that may be encountered during a loss of offsite power. Operators can manually open the PORVs to reduce PCS pressure in the event of a Steam Generator Tube Rupture (SGTR) with offsite power unavailable.

The PORVs 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.

BASES

BACKGROUND (continued)

If preferred during normal plant operation when PCS temperature is at or above 430°F and the PORV block valves are open, the PORVs may also function as an automatic overpressure device and limits challenges to the safety valves. Although the PORVs act as an overpressure device for operational purposes, safety analyses do not take credit for PORV actuation, but do take credit for the safety valves. Since the pressurizer safety valves provide the necessary automatic protection against excessive PCS pressure, automatic actuation of the PORVs is not required to be OPERABLE and the PORVs and their block valves are normally maintained in the closed position.

The PORVs also provide Low Temperature Overpressure Protection (LTOP) during heatup and cooldown. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," 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. The possibility of a small break LOCA through the PORV is reduced when the PORV flow path is OPERABLE and the PORV opening setpoint is established to be reasonably remote from expected transient challenges. The possibility is further 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. However, technical findings and regulatory analysis discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants," have determined that maintaining the requirements for PORVs and block valves in the technical specifications can increase the reliability of these components and provide assurance they will function as required and that operating experience has shown these components to be important to public health and safety.

Pressurizer PORVs satisfy Criterion 4 of 10 CFR 50.36(c)(2).

BASES

LCO

The LCO requires each PORV and its associated block valve to be OPERABLE. The block valve is required to be OPERABLE so it may be used to isolate the flow path of an inoperable PORV or, unisolate the flow path of an OPERABLE PORV. Thus, a block valve is considered OPERABLE if it is capable of being cycled in the open and close direction.

The PORV is required to be OPERABLE to provide PCS pressure control and maintain PCS integrity. For a PORV, OPERABILITY means the valve is capable of being cycled in the open and close direction.

APPLICABILITY

With a PORV in the "CLOSED" position in MODES 1 and 2, and MODE 3 with all PCS cold leg temperatures ≥ 430°F, the PORV and its block valve are required to be OPERABLE to limit PCS leakage through the PORV flow path, and to be available for manual operation to mitigate abnormal transients which may be initiated from these MODES and condition.

With a PORV in the "AUTO" position in MODES 1 and 2, and MODE 3 with all PCS cold leg temperatures ≥ 430°F, 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 PCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the steam generators 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 with PCS cold leg temperatures < 430°F because the core input energy is reduced, but the PCS pressure is high. Therefore, this LCO is applicable in MODES 1 and 2, and MODE 3 with all PCS cold leg temperatures ≥ 430°F.

The LCO is not applicable in MODE 3 with any PCS cold leg temperatures < 430°F 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 when any PCS cold leg temperatures are < 430°F, and in MODES 4, 5, and MODE 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

BASES

ACTIONS

The ACTIONS are modified by two Notes. Note 1 clarifies that each pressurizer PORV is treated as a separate entity, each with separate Completion Times (i.e., the Completion Time is on a component basis). Note 2 is an exception to LCO 3.0.4. The exception for LCO 3.0.4 permits entry into MODES 1, 2, and 3 to perform cycling of the PORV to verify their OPERABLE status.

A.1 and A.2

If one PORV is inoperable it must either be isolated, by closing the associated block valve, or restored to OPERABLE status. The Completion Time of 1 hour is reasonable based on the small potential that the PORVs will be required to function 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, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. Mechanical system LCOs typically provide a 72 hour Completion Time under conditions when a required system can perform its required safety function, but may not be able to do so assuming an additional failure. When operating in accordance with the Required Actions of an LCO Condition, it is not necessary to be able to cope with an additional single failure

B.1 and B.2

If one block valve is inoperable, then it must be restored to OPERABLE status, or the associated PORV placed in manual control. Placing a PORV in manual control is accomplished by placing the PORV hand switch in the "CLOSE" position. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable based on the small potential that the PORVs will be required to function during this time period and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status.

BASES

ACTIONS

B.1 and B.2 (continued)

Mechanical system LCOs typically provide a 72 hour Completion Time under conditions when a required system can perform its required safety function, but may not be able to do so assuming an additional failure. When operating in accordance with the Required Actions of an LCO Condition, it is not necessary to be able to cope with an additional single failure.

The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition A since the PORVs are not capable of automatically mitigating an overpressure event when placed in manual control. If the block valve is restored within the Completion Time of 72 hours, the PORV is restored to OPERABLE status.

C.1 and C.2

If more than one PORV is inoperable, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing the associated block valves and restoring at least one PORV to OPERABLE status within 2 hours. The Completion Time of 1 hour is reasonable based on the small potential that the PORVs will be required to function during this time period, and provides the operator time to correct the situation. If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition A with the time clock started at the original declaration of having two PORVs inoperable.

D.1 and D.2

If two block valves are inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour or place the associated PORVs in manual control and restore at least one block valve to OPERABLE status within 2 hours and the remaining block valve in 72 hours. The Completion Time of 1 hour to either restore the block valves or place the associated PORVs in manual control is reasonable based on the small potential that the PORVs will be required to function during this time period, and provides the operator time to correct the situation.

BASES

ACTIONS (continued)

E.1

If the Required Actions and associated Completion Times are not met, then the plant must be brought to a stable condition which minimizes the potential for transients affecting the PCS. The plant must be brought to at least MODE 3 within 6 hours. With one or two PORVs or block valves inoperable, exiting the MODE of Applicability (i.e., MODE 3 with any PCS cold leg temperature < 430°F) may not be desirable since below 430°F the PORVs and their associated block valves are required to support LTOP operations (LCO 3.4.12). Although LCO 3.0.4 would allow entry into LCO 3.4.12, reducing PCS temperature below 430°F may not be prudent since below 430°F the PORVs are credited in the safety analysis to protect the PCS from an inadvertent overpressure event. At or above 430°F, the PORVs are not credited in the safety analysis and thus have no safety function. If practical, the inoperable PORVs or block valves should be restored to an OPERABLE status while the PCS is above 430°F to avoid entering a plant condition where the PORVs are required for LTOP. If necessary, LCO 3.0.4 would allow the plant to be placed in MODE 5 to facilitate repairs. In this plant condition, overpressure protection may be provided by establishing the required vent path specified in LCO 3.4.12.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. In MODE 3 with any PCS cold leg temperature < 430°F, and MODES 4 and 5 and MODE 6 with the reactor vessel head on, maintaining PORV OPERABILITY is required by LCO 3.4.12.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that it can be opened and closed if necessary. The basis for the Frequency of "prior to entering MODE 4 from MODE 5 if not performed in the previous 92 days" reflects the importance of not routinely cycling the block valves during the period when the PCS is pressurized since this practice may result in the associated PORV being opened by the increase inlet pressure to the PORV. The "92 days" portion of the Frequency is consistent with the testing frequency stipulated by ASME Section XI as modified by the Cold Shutdown Testing Basis used in support of the second 120 month interval of the Inservice Valve Testing Program which only requires the block valves to be cycled during Cold Shutdown conditions. 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 primary coolant pressure. If a block valve is open and its associated PORV was stuck open, the OPERABILITY of the block valve is of importance because closing the block valve is necessary to isolate the stuck opened PORV.

SR 3.4.11.2

SR 3.4.11.2 requires complete cycling of each PORV. PORV cycling demonstrates its function and is performed when the PCS temperature is > 200°F. Stroke testing of the PORVs above 200°F is desirable since it closer simulates the temperature and pressure environmental effects on the valves and thus represents a better test condition for assessing PORV performance under normal plant conditions. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

None

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls PCS pressure at low temperatures so the integrity of the Primary Coolant Pressure Boundary (PCPB) is not compromised by violating the Pressure and Temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting PCPB component requiring such protection. LCO 3.4.3, "PCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 requirements during the LTOP MODES.

The toughness of the reactor vessel material decreases at low temperatures. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). PCS 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 PCS is water solid, which occurs only while shutdown. Under that condition, a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the PCS P/T limits by a significant amount could cause brittle fracture of the reactor vessel. LCO 3.4.3 requires administrative control of PCS pressure and temperature during heatup and cooldown to prevent exceeding the P/T limits.

This LCO provides PCS overpressure protection by limiting coolant injection capability and requiring adequate pressure relief capacity. Limiting coolant injection capability requires all High Pressure Safety Injection (HPSI) pumps be incapable of injection into the PCS when any PCS cold leg temperature is < 300°F. The pressure relief capacity requires either two OPERABLE redundant Power Operated Relief Valves (PORVs) or the PCS depressurized and a PCS vent of sufficient size. One PORV or the PCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

BASES

BACKGROUND (continued)

With limited coolant injection capability, the ability to provide core coolant addition is restricted. The LCO does not require the chemical and volume control system to be deactivated or the Safety Injection Signals (SIS) blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the chemical and volume control system can provide adequate flow via the makeup control valve. If conditions require the use of an HPSI pump for makeup in the event of loss of inventory, then a pump can be made available through manual actions.

The LTOP System for pressure relief consists of two PORVs with temperature dependent lift settings or a PCS vent of sufficient size. Two PORVs are required for redundancy. One PORV has adequate relieving capability to prevent overpressurization for the allowed coolant injection capability.

PORV Requirements

As designed for the LTOP System, an "open" signal is generated for each PORV if the PCS pressure approaches a limit determined by the LTOP actuation logic. The actuation logic monitors PCS pressure and cold leg temperature to determine when the LTOP overpressure setting is approached. If the indicated pressure meets or exceeds the calculated value, a PORV is opened.

The LCO presents the PORV setpoints for LTOP by specifying Figure 3.4.12-1, "LTOP Setpoint Limit." 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 and the valve closed. The pressure continues to decrease below the reset pressure as the valve closes.

BASES

BACKGROUND (continued)

PCS Vent Requirements

Once the PCS is depressurized, a vent exposed to the containment atmosphere will maintain the PCS at containment ambient pressure in an PCS 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 injection or heatup transient and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

Reference 3 has determined that any vent path capable of relieving 167 gpm at a PCS pressure of 315 psia is acceptable. The 167 gpm flow rate is based on an assumed charging imbalance due to interruption of letdown flow with three charging pumps operating, a 40°F per hour PCS heatup rate, a 60°F per hour pressurizer heatup rate, and an initially depressurized and vented PCS. Neither HPSI pump nor Primary Coolant Pump (PCP) starts need to be assumed with the PCS initially depressurized, because LCO 3.4.12 requires both HPSI pumps to be incapable of injection into the PCS and LCO 3.4.7, "PCS Loops-MODE 5, Loops Filled," places restrictions on starting a PCP.

The pressure relieving ability of a vent path depends not only upon the area of the vent opening, but also upon the configuration of the piping connecting the vent opening to the PCS. A long, or restrictive piping connection may prevent a larger vent opening from providing adequate flow, while a smaller opening immediately adjacent to the PCS could be adequate. The areas of multiple vent paths cannot simply be added to determine the necessary vent area.

The following vent path examples are acceptable:

1. Removal of a steam generator primary manway;
2. Removal of the pressurizer manway;
3. Removal of a PORV or pressurizer safety valve;
4. Both PORVs and associated block valves open; and
5. Opening of both PCS vent valves MV-PC514 and MV-PC515.

BASES

BACKGROUND (continued)

Reference 4 determined that venting the PCS through MV-PC514 and MV-PC515 provided adequate flow area. The other listed examples provide greater flow areas with less piping restriction and are therefore acceptable. Other vent paths shown to provide adequate capacity could also be used. The vent path(s) must be above the level of reactor coolant, to prevent draining the PCS.

One open PORV provides sufficient flow area to prevent excessive PCS pressure. However, if the PORVs are elected as the vent path, both valves must be used to meet the single failure criterion, since the PORVs are held open against spring pressure by energizing the operating solenoid.

When the shutdown cooling system is in service with MO-3015 and MO-3016 open, additional overpressure protection is provided by the relief valves on the shutdown cooling system. References 5 and 6 show that this relief capacity will prevent the PCS pressure from exceeding its pressure limits during any of the above mentioned events.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 7) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits during shutdown. In MODES 1 and 2, and in MODE 3 with all PCS cold leg temperature at or exceeding 430°F, the pressurizer safety valves prevent PCS pressure from exceeding the Reference 1 limits. Below 430°F, overpressure prevention falls to the OPERABLE PORVs or to a depressurized PCS and a sufficiently sized PCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP System should be re-evaluated to ensure its functional requirements can still be satisfied using the PORV method or the depressurized and vented PCS condition.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

Transients that are capable of overpressurizing the PCS are categorized as either mass injection or heatup transients

Mass Injection Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

Heatup Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of Shutdown Cooling (SDC); or

c. PCP startup with temperature asymmetry within the PCS or between the PCS and steam generators.

Rendering both HPSI pumps incapable of injection is required during the LTOP MODES to ensure that mass injection transients beyond the capability of the LTOP overpressure protection system, do not occur. The Reference 3 analyses demonstrate that either one PORV or the PCS vent can maintain PCS pressure below limits when three charging pump are actuated. Thus, the LCO prohibits the operation of both HPSI pumps and does not place any restrictions on charging pump operation.

Fracture mechanics analyses were used to establish the applicable temperature range for the LTOP LCO as below 430°F. At and above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 2.192 E19 nvt.

BASES

APPLICABLE SAFETY ANALYSES (continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the setpoint curve specified in Figure 3.14.12-1 of the accompanying LCO. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient. The valve qualification process considered pressure overshoot and undershoot beyond the PORV opening and closing setpoints, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensure the Reference 1 limits will be met.

The PORV setpoints will be re-evaluated for compliance when the P/T limits are revised. 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.

PCS Vent Performance

With the PCS depressurized, analyses show the required vent size is capable of mitigating the limiting allowed LTOP overpressure transient. In that event, this size vent maintains PCS pressure less than the maximum PCS pressure on the P/T limit curve.

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

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

BASES

LCO

This LCO is required to ensure that the LTOP System is OPERABLE. The LTOP System is OPERABLE when both HPSI pumps are incapable of injecting into the PCS 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 limits as a result of an operational transient.

To limit the coolant injection capability, LCO 3.4.12.a requires both HPSI pumps be incapable of injecting into the PCS. LCO 3.4.12.a is modified by two Notes. Note 1 only requires both HPSI pumps to be incapable of injecting into the PCS when any PCS cold leg temperature is < 300°F. When all PCS cold leg temperatures are ≥ 300°F, a start of both HPSI pumps in conjunction with a charging/letdown imbalance will not cause the PCS pressure to exceed the 10 CFR 50 Appendix G limits. Thus, a restriction on HPSI pump operation when all PCS cold leg temperatures are ≥ 300°F is not required. Note 2 is provided to assure that this LCO does not cause hesitation in the use of a HPSI pump for PCS makeup if it is needed due to a loss of shutdown cooling or a loss of PCS inventory.

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

a. Two OPERABLE PORVs; or

b. The PCS depressurized and vented.

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set consistent with Figure 3.4.12-1 in the accompanying LCO and testing has proven its ability to open at that setpoint, and motive power is available to the valve and its control circuit.

A PCS vent is OPERABLE when open with an area capable of relieving ≥ 167 gpm at a PCS pressure of 315 psia.

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

BASES

APPLICABILITY

This LCO is applicable in MODE 3 when the temperature of any PCS cold leg is < 430°F, in MODES 4 and 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits at and above 430°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1 and 2, and MODE 3 with all PCS cold leg temperatures ≥ 430°F.

Low temperature overpressure prevention is most critical during shutdown when the PCS is water solid, and a mass addition or a heatup transient can cause a very rapid increase in PCS pressure with little or no time available for operator action to mitigate the event.

ACTIONS

A Note has been added in the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS, even though the ACTIONS may eventually require a plant shutdown. The intent of this exception is to allow the plant to enter the LTOP MODE with an inoperable PORV from MODES 1 and 2, and MODE 3 with all PCS cold leg temperature ≥ 430°F, to facilitate valve repairs. This exception is acceptable since the Required Actions provide the appropriate compensatory measures commensurate with PORV inoperabilites.

<u>A.1</u>

With one or two HPSI pumps capable of injecting into the PCS, overpressurization is possible.

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

BASES

ACTIONS (continued)

B.1

With one required PORV inoperable and pressurizer water level ≤ 57%, the required PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two valves are 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 only one PORV being required to mitigate an overpressure transient, the likelihood of an active failure of the remaining valve path during this time period being very low, and that a steam bubble exists in the pressurizer. Since the pressure response to a transient is greater if the pressurizer steam space is small or if the PCS is solid, the Completion Time for restoration of a PORV flow path to service is shorter. The maximum pressurizer level at which credit can be taken for having a bubble (57%, which provides about 700 cubic feet of steam space) is based on judgement rather than by analysis. This level provides the same steam volume to dampen pressure transients as would be available at full power. This steam volume provides time for operator action (if the PORVs failed to operate) in the interval between an inadvertent SIS and PCS pressure reaching the 10 CFR 50, Appendix G pressure limit. The time available for action would depend upon the existing pressure and temperature when the inadvertent SIS occurred.

C.1

The consequences of operational events that will overpressurize the PCS are more severe at lower temperature (Ref. 8). With the pressurizer water level > 57%, less steam volume is available to dampen pressure increases resulting from an inadvertent mass injection or heatup transients. Thus, with one required PORV inoperable and the pressurizer water level > 57%, the Completion Time to restore the required PORV to OPERABLE status is 24 hours.

The 24 hour Completion Time to restore the required PORV to OPERABLE status when the pressurizer water level is > 57%, which usually occurs in MODE 5 or in MODE 6 when the vessel head is on, is a reasonable amount of time to investigate and repair PORV failures without a lengthy period with only one PORV OPERABLE to protect against overpressure events.

BASES

ACTIONS (continued)

D.1

If two required PORVs are inoperable, or if the Required Actions and the associated Completion Times are not met, or if the LTOP System is inoperable for any reason other than Condition A, B, or C, the PCS must be depressurized and a vent established within 8 hours. The vent must be sized to provide a relieving capability of ≥ 167 gpm at a pressure of 315 psia which ensures the flow capacity is greater than that required for the worst case mass injection transient reasonable during the applicable MODES. This action protects the PCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time of 8 hours to depressurize and vent the PCS 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 operator attention and administrative requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

To minimize the potential for a low temperature overpressure event by limiting the mass injection capability, both HPSI pumps are verified to be incapable of injecting into the PCS. The HPSI pumps are rendered incapable of injecting into the PCS by means that assure that a single event cannot cause overpressurization of the PCS due to operation of the pump. Typical methods for accomplishing this are by pulling the HPSI pump breaker control power fuses, racking out the HPSI pump motor circuit breaker, or closing the manual discharge valve.

SR 3.4.12.1 is modified by a Note which only requires the SR to be met when complying with LCO 3.4.12.a. When all PCS cold leg temperature are ≥ 300°F, a start of both HPSI pumps in conjunction with a charging/letdown imbalance will not cause the PCS pressure to exceed the 10 CFR 50 Appendix G limits. Thus, this SR is only required when any PCS cold leg temperature is reduced to less than 300°F.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.2

SR 3.4.12.2 requires a verification that the required PCS vent, capable of relieving ≥ 167 gpm at a PCS pressure of 315 psia, is OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked open; or

b. Once every 31 days for a valve that is locked open.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance need only be performed if vent valves are being used to satisfy the requirements of this LCO. This Surveillance does not need to be performed for vent paths relying on the removal of a steam generator primary manway cover, pressurizer manway cover, safety valve or PORV since their position is adequately addressed using administrative controls and the inadvertent reinstallation of these components is unlikely. The Frequencies consider operating experience with mispositioning of unlocked and locked vent valves, respectively.

SR 3.4.12.3

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, motor operated valve. 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.4

Performance of a CHANNEL FUNCTIONAL TEST is required every 31 days. 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 Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. PORV actuation could depressurize the PCS 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 any PCS cold leg temperature to < 430°F. This Note allows a discrete period of time to perform the required test without delaying entry into the MODE of Applicability for LTOP. This option may be exercised in cases where an unplanned shutdown below 430°F is necessary as a result of a Required Action specifying a plant shutdown, or other plant evolutions requiring an expedited cooldown of the plant. The test must be performed within 12 hours after entering the LTOP MODES.

SR 3.4.12.5

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the entire channel so that it responds and the valve opens within the required LTOP range and with accuracy to known input.

The 18 month Frequency considers operating experience with equipment reliability and is consistent with the typical refueling outage schedule.

BASES

REFERENCES

1. 10 CFR 50, Appendix G
2. Generic Letter 88-11
3. CPC Engineering Analysis, EA-A-PAL-92095-01
4. CPC Engineering Analysis, EA-TCD-91-01-01
5. CPC Engineering Analysis, EA-PAL-89-040-1
6. CPC Corrective Action Document, A-PAL-91-011
7. FSAR, Section 7.4
8. Generic Letter 90-06

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.13 PCS Operational LEAKAGE

BASES

BACKGROUND

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

During plant life, the joint and valve interfaces can produce varying amounts of primary coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the PCS 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.

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

The safety significance of PCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring primary 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 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 PCS LEAKAGE detection.

This LCO deals with protection of the Primary Coolant Pressure Boundary (PCPB) 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 Loss Of Coolant Accident (LOCA).

BACKGROUND (continued)

As defined in 10 CFR 50.2, the PCPB includes all those pressure-containing components, such as the reactor pressure vessel, piping, pumps, and valves, which are:

(1) Part of the primary coolant system, or

(2) Connected to the primary coolant system, up to and including any and all of the following:

(i) The outermost containment isolation valve in system piping which penetrates the containment,

(ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate the containment,

(iii) The pressurizer safety valves and PORVs.

APPLICABLE SAFETY ANALYSES

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 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 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a Main Steam Line Break (MSLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a Steam Generator Tube Rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 2) analysis for SGTR assumes the contaminated secondary fluid is released via the Main Steam Safety Valves and Atmospheric Dump Valves. The 1 gpm primary to secondary LEAKAGE is inconsequential, relative to the dose contribution from the affected SG.

The MSLB is more limiting for site radiation releases. The safety analysis for the MSLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the MSLB accident are well within the guidelines defined in 10 CFR 100.
PCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

PCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE from within the PCPB is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in increased LEAKAGE. Violation of this LCO could result in continued degradation of the PCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

As defined in Section 1.0, pressure boundary LEAKAGE is "LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an PCS component body, pipe wall, or vessel wall."

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE from within the PCPB 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 PCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE from within the PCPB is allowed because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the PCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically located sources which is known not to adversely affect the OPERABILITY of required leakage detection systems, but does not include pressure boundary LEAKAGE or controlled Primary Coolant Pump (PCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

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

LCO (continued)

d. Primary to Secondary LEAKAGE through Any One SG

The 432 gallons per day limit on primary to secondary LEAKAGE through any one SG ensures the total primary to secondary LEAKAGE through both SGs produces acceptable offsite doses in the MSLB accident analysis. In addition, the LEAKAGE limit also ensures that SG integrity is maintained in the event of an MSLB or under LOCA conditions. Violation of this LCO could exceed the offsite dose limits for this accident analysis. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

APPLICABILITY

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

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

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the PCPB.

B.1 and B.2

If any pressure boundary LEAKAGE from within the PCPB exists or if unidentified, identified, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. 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 PCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying PCS LEAKAGE to be within the LCO limits ensures the integrity of the PCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an PCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of a PCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The PCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. Therefore, this SR is modified by a Note which states that the SR is not required to be performed in MODES 3 and 4, until 12 hours of steady state operation have elapsed.

Steady state operation is required to perform a proper water inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met only when steady state is established. For PCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable PCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and PCP seal leakoff.

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 3.4.15, "PCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be performed during steady state operation.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. Regulatory Guide 1.45, May 1973
2. FSAR, Section 14.15

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.14 PCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

The Reactor Safety Study (RSS), WASH-1400 (Ref. 1), identified a special class of Loss of Coolant Accidents (LOCAs) where the accident is initiated by the failure of check valves which seperate the high pressure Primary Coolant System (PCS) from lower pressure systems connected to the PCS. This check valve failure could cause overpressurization and rupture of the lower pressure piping and result in a LOCA that bypasses containment. With the containment bypassed, the leakage would not be available for recirculation and when the Safety Injection Refueling Water Tank (SIRWT) emptied core cooling would be lost. This event has become known as "Event V."

When pressure isolation is provided by two in-series check valves and failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important to safety, they should be tested periodically to ensure low probability of gross failure. Periodic examination of check valves must be undertaken to verify that each valve is seated properly and functioning as a pressure isolation device. The testing will reduce the overall risk of an inter-system LOCA. The testing may be accomplished by direct volumetric leakage measurement or by other equivalent means capable of demonstrating that leakage limits are not exceeded. The PCS PIV LCO allows PCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety. The PIV leakage limit applies to each individual valve. Leakage through both PIVs in series in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "PCS Operational LEAKAGE." This is true during operation only when the loss of PCS mass through two valves in series is determined by a water inventory balance (SR 3.4.13.1).

A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not PCS operational LEAKAGE if the other is leaktight.

BASES

BACKGROUND (continued)

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. Therefore, this specification also addresses the potential for overpressurization of the low pressure piping in the Shutdown Cooling (SDC) system caused by the inadvertent opening of the SDC suction valves (MO-3015 and MO-3016) when the PCS pressure is above the design pressure of the SDC System. The leakage limit is an indication that the PIVs between the PCS and the connecting systems are degraded or degrading. PIV leakage or inadvertent valve positioning could lead to overpressure of the low pressure piping or components. Failure consequences could be a LOCA outside of containment, which is an unanalyzed condition that could degrade the ability for low pressure injection.

PIVs are provided to isolate the PCS from the following systems:

a. Shutdown Cooling System; and

b. Safety Injection System.

The PIVs which are required to be leak tested are listed in Table B 3.4.14-1.

Violation of this LCO could result in overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 1 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of low pressure piping outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the Primary Coolant Pressure Boundary (PCPB), and the subsequent pressurization of the lower pressure piping downstream of the PIVs from the PCS. Overpressurization failure of the lower pressure piping would result in a LOCA outside containment and subsequent risk of core melt.

Reference 2 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

PCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2).

BASES

LCO

PCS PIV leakage is identified LEAKAGE into closed systems connected to the PCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that corrective action must be taken. The PIVs which are required to be leak tested are listed in Table 3.4.14-1.

The LCO PIV leakage limit is a maximum of 5 gpm. Reference 3 permits leakage testing at a lower pressure differential than that between maximum PCS pressure and the normal pressure of the connected system during PCS operation (the maximum pressure differential). The observed leakage rate must be corrected to the maximum pressure differential, assuming leakage is directly proportional to the square root of pressure differential.

The LCO also requires the SDC suction valve interlocks to be OPERABLE in order to prevent the inadvertent opening of the SDC suction valves when PCS pressure is above the 300 psig design pressure of the SDC suction piping. When PCS pressure is ≥ 280 psia as sensed by the pressurizer narrow range pressure channels, an inhibit signal is placed on the control circuit for the SDC suction valves which prevents the valves from opening and thus avoiding a potential overpressurization event of the SDC piping. For the SDC suction valve interlocks to be OPERABLE, two channels of pressurizer narrow range pressure instruments must be capable of providing an open inhibit signal to their respective isolation valve.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the PCS is pressurized. In MODE 4, the requirements of this LCO are not required when in, or during the transition to or from, the SDC mode of operation since these evolutions are performed when PCS pressure is less than the limiting design pressure of the systems addressed by this specification.

In MODES 5 and 6, leakage limits are not provided because the lower primary coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

BASES

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based on the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

Required Action A.1 requires that isolation with one valve must be performed within 4 hours whenever one or more flow paths with leakage from one or more PIVs is not within limits. Four hours provides time to reduce leakage in excess of the allowable limit or to isolate the flow path if leakage cannot be reduced while restricting operation with leaking isolation valves. Required Action A.1 is modified by a Note stating that the valves used for isolation must meet the same leakage requirement as the PIVs and must be in the PCPB or the high pressure portion of the system.

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This time frame considers the time required to complete this action and the low probability of a second valve failing during this period.

B.1 and B.2

If leakage cannot be reduced or if the affected system can not be isolated within the specified 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 MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

C.1

The inoperability of the SDC suction valve interlocks renders the SDC suction isolation valves incapable of preventing an inadvertent opening of the valves at PCS pressures in excess of the SDC systems design pressure. If the SDC suction valve interlocks are inoperable, operation may continue as long as the suction penetration is closed by at least one closed deactivated valve within 4 hours. This action accomplishes the purpose of the interlock. The 4 hour Completion Time provides time to accomplish the action and restricts operation with an inoperable interlock.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

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

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

Testing is to be performed every 9 months whenever the plant has been in MODE 5 for 7 days or more, but may be extended up to a maximum of 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g), as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI, and is based on the need to perform the Surveillance under conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

SR 3.4.14.1 is modified by three Notes. Note 1 states that the SR is only required to be performed in MODES 1 and 2. Entry into MODES 3 and 4 is allowed to establish the necessary differential pressure and stable conditions to allow performance of this surveillance.

Note 2 further restricts the PIV leakage rate acceptance criteria by limiting the reduction in margin between the measured leakage rate and the maximum permissible leakage rate by 50% or greater. Reductions in margin by 50% or greater may be indicative of PIV degradation and warrant inspection or additional testing. Thus, leakage rates less than 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

Note 3 limits the minimum test differential pressure to 150 psid during performance of PIV leakage testing.

SR 3.4.14.2

Verifying that the SDC suction valve interlocks are OPERABLE ensures that PCS pressure will not pressurize the SDC system beyond 125% of its design pressure of 300 psig. The interlock setpoint that prevents the valves from being opened is set so the actual PCS pressure must be < 280 psia to open the valves. This setpoint ensures the SDC design pressure will not be exceeded and the SDC relief valves will not lift. The narrow range pressure transmitters that provide the SDC suction valve interlocks are sensed from the pressurizer. Due to the elevation differences between these narrow range pressure transmitter calibration points and the SDC suction piping, the pressure in the SDC suction piping will be higher than the indicated pressurizer pressure. Due to this pressure difference, the SDC suction valve interlocks are conservatively set at or below 280 psia to ensure that the 300 psig (315 psia) design pressure of the suction piping is not exceeded. The 18 month Frequency is based on the need to perform these Surveillances under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.3

This SR requires a verification that the four Low Pressure Safety Injection (LPSI) check valves (CK-3103, CK-3118, CK-3133 and CK-3148) in the SDC flow path reclose after stopping SDC flow. Performance of this SR is necessary to ensure the LPSI check valves are closed to prevent overpressurization of the LPSI subsystem from the High Pressure Safety Injection (HPSI) subsystem. Overpressurization of the LPSI piping could occur if the LPSI check valves were not closed upon the receipt of a Safety Injection Signal and PCS pressure remained relatively high (e.g., during a small break LOCA). In this case, the higher pressure water from the discharge of the HPSI pumps could cause the lower pressure LPSI piping to exceed its design pressure. This event could result in a loss of emergency core cooling water outside containment which reduces the volume of water available for recirculation from the containment sump (Ref. 4).

SR 3.4.14.3 is required to be performed on a Frequency of "prior to entering MODE 2 whenever the LPSI check valves have been used for SDC." This ensures the LPSI check valves are closed whenever they have been opened for SDC operations prior to a reactor startup. The SR is modified by a Note which states that the surveillance is only required to be performed in MODES 1 and 2. Thus, entry into MODES 3 and 4 is allowed to establish the necessary differential pressure and to establish stable conditions to allow performance of this surveillance.

REFERENCES

1. WASH-1400 (NUREG-75/014), Appendix V, October 1975
2. NUREG-0677, May 1980
3. ASME, Boiler and Pressure Vessel Code, Section XI
4. Letter from Consumers Power Company to D.M. Crutchfield (NRC) Requesting a Change to the Palisades Plant Technical Specification, dated July 29, 1982

BASES

TABLE B 3.4.14-1 (page 1 of 1)
Required PCS Pressure Isolation Valves

System	Valve No.
High Pressure Safety Injection	
Loop 1A, Cold Leg	CK - 3101 CK - 3104
Loop 1B, Cold Leg	CK - 3116 CK - 3119
Loop 2A, Cold Leg	CK - 3131 CK - 3134
Loop 2B, Cold Leg	CK - 3146 CK - 3149
Low Pressure Safety Injection	
Loop 1A, Cold Leg	CK - 3103
Loop 1B, Cold Leg	CK - 3118
Loop 2A, Cold Leg	CK - 3133
Loop 2B, Cold Leg	CK - 3148

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.15 PCS Leakage Detection Instrumentation

BASES

BACKGROUND

The Palisades Nuclear Plant design criteria (Ref. 1) require means for detecting and, to the extent practical, identifying the location of the source of PCS LEAKAGE.

Leakage detection instrumentation must have the capability to detect significant Primary Coolant Pressure Boundary (PCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 gpm to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump, which is used to collect unidentified LEAKAGE, is instrumented with level transmitters providing sump level indication in the control room. The sensitivity of these instruments is acceptable for detecting increases in unidentified LEAKAGE.

The primary coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Primary 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. An instrument sensitivity capable of detecting a 100 cm^3/min leak in 45 minutes based on 1% failed fuel is practical for the leakage detection instrument (Ref. 2). Radioactivity detection is included for monitoring gaseous activities because of its sensitivity to PCS LEAKAGE.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Humidity detectors are capable of detecting a 10% change in humidity which would result from approximately 150 gallons of primary water leakage (Ref. 2).

BASES

BACKGROUND (continued)

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump and condensate flow from the containment air coolers. Humidity level monitoring is considered most useful as an indirect indication to alert the operator to a potential problem.

The containment air cooler design includes a sump with a drain, a liquid level switch, and an overflow path. Normally, very little water will be condensed from the containment atmosphere and the small amount of condensate will easily flow out through the sump drain. If leakage through the drain is greater than 25 gpm, the level in the sump will rise to the liquid level switch (approximately 6 inches from the bottom of the sump) and triggers an alarm in the control room to alert the operators of the excessive cooling coil drainage. Excessive drain water flow from the coils is indicative of a service water leak, steam leak, or a primary coolant system leak. A steam leak or primary coolant leak would be accompanied by an increase in the containment atmosphere humidity which would be detected by the containment humidity sensors and displayed in the control room. Since excessive containment air cooler drainage may be attributed to causes other than PCS LEAKAGE, an evaluation of PCS LEAKAGE should be confirmed using diverse instrumentation required by this specification.

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

BASES

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 system sensitivities are described in the FSAR (Ref. 2). Multiple instrument locations are utilized, if needed, to ensure the transport delay time of the LEAKAGE from its source to an instrument location is acceptable.

The safety significance of PCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring PCS 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.

PCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2).

LCO

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

The LCO is satisfied when monitors of diverse measurement means are available. Thus, a combination which includes one instrument channel from each of any three of the following; containment sump level indication, gaseous activity monitor, containment air cooler condensate level switch, or containment humidity monitor provides an acceptable minimum. For the containment air cooler condensate level switch only an operating containment air cooler may be relied upon to fulfill the LCO requirements for an OPERABLE leakage detection instrument.

APPLICABILITY

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

In MODE 5 or 6, the temperature is ≤ 200°F and pressure is maintained low or at atmospheric pressure.

BASES

APPLICABILITY (continued)

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

The ACTIONS are modified by a Note that indicates the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one or two required leak detection instrument channels are inoperable. This allowance is provided because other instrumentation is available to monitor for PCS LEAKAGE.

A.1 and A.2

If one or two required leak detection instrument channels are inoperable, a periodic surveillance for PCS 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.

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per. . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.4.15 A.1 must be initially performed within 24 hours without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

Restoration of the required instrument channels to an OPERABLE status is required to regain the function in a Completion Time of 30 days after the instrument's failure. This time is acceptable considering the frequency and adequacy of the PCS water inventory balance required by Required Action A.1.

B.1 and B.2

If the Required Action 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.

BASES

ACTIONS (continued)

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1, SR 3.4.15.2, and SR 3.4.15.3

These SRs require the performance of a CHANNEL CHECK for each required containment sump level indicator, containment atmosphere gaseous activity monitor, and containment atmosphere humidity monitor. 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 3.4.15.4

SR 3.4.15.4 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment air cooler condensate level switch. Since this instrumentation does not include control room indication of flow rate, a CHANNEL CHECK is not possible. The test ensures that the level switch can perform its function in the desired manner. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Frequency of 18 months is a typical refueling cycle (performance of the test is only practical during a plant outage) and considers instrument reliability. Operating experience has shown this Frequency is acceptable for detecting degradation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.15.5, SR 3.4.15.6, and SR 3.4.15.7

These SRs require the performance of a CHANNEL CALIBRATION for each required containment sump level, containment atmosphere gaseous activity, and containment atmosphere humidity channel. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Operating experience has shown this Frequency is acceptable.

REFERENCES

1. FSAR, Section 5.1.5
2. FSAR, Sections 4.7 and 6.3

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.16 PCS Specific Activity

BASES

BACKGROUND

10 CFR 100.11 specifies the maximum dose to the whole body and the thyroid an individual at the site boundary can receive for 2 hours during an accident. The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 guideline limits during analyzed transients and accidents.

The PCS specific activity LCO limits the allowable concentration level of radionuclides in the primary coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a Steam Generator Tube Rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for an SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following an SGTR accident. The SGTR safety analysis (Ref. 1) assumes the specific activity of the primary coolant at the LCO limits and an existing primary coolant Steam Generator (SG) tube leakage rate of 1 gpm. The analysis also assumes a reactor trip and a turbine trip at the same time as the SGTR event.

The analysis for the SGTR accident establishes the acceptance limits for PCS specific activity. Reference to this analysis is used to assess changes to the facility that could affect PCS specific activity as they relate to the acceptance limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The rise in pressure in the ruptured SG causes radioactive contaminated steam to discharge to the atmosphere through the atmospheric dump valves or the main steam safety valves. The atmospheric discharge stops when the affected SG is isolated below approximately 525°F. The unaffected SG removes core decay heat by venting steam until Shutdown Cooling conditions are reached.

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the 10 CFR 100 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limit of 40 μCi/gm for more than 48 hours.

This is acceptable because of the low probability of an SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

PCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the primary coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 1) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in primary coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

BASES

APPLICABILITY

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

For operation in MODE 3 with PCS 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 corresponding to the primary coolant temperature is below the lift 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 4 hours must be taken to demonstrate the limit 40 μCi/gm is not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample.

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per. . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.4.16 A.1 must be initially performed within 4 hours without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

Sampling must continue for trending. The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours.

The Completion Time of 48 hours is required if the limit violation resulted from normal iodine spiking.

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to, power operation.

BASES

ACTIONS (continued)

B.1

If a Required Action and associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is 40 μCi/gm or above, or with the gross specific activity in excess of the allowed limit, the plant must be placed in a MODE in which the requirement does not apply.

The change within 6 hours to MODE 3 with PCS average temperature < 500°F lowers the saturation pressure of the primary 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 6 hours is required to reach MODE 3 below 500°F from full power conditions and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

The Surveillance requires performing a gamma isotopic analysis as a measure of the gross specific activity of the primary coolant at least once per 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with PCS average temperature at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance 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 2 hours and 6 hours after a power change of ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results. This SR is modified by a Note which states that the SR is only required to be performed in MODE 1.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 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 required to be performed within 31 days after 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. FSAR, Section 14.15

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Safety Injection Tanks (SITs)

BASES

BACKGROUND

The functions of the four SITs are to supply water to the reactor vessel during the blowdown phase of a Loss of Coolant Accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Primary Coolant System (PCS) makeup for a small break LOCA.

The blowdown phase of a LOCA is the initial period of the transient during which the PCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the primary coolant. The blowdown phase of the transient ends when the PCS pressure falls to a value approaching that of the containment atmosphere.

The refill phase of a LOCA follows immediately after the primary coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of the SITs' inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of Safety Injection (SI) water.

The SITs are pressure vessels partially filled with borated water and pressurized with nitrogen gas (Ref. 2). The SITs are passive components, since no operator or control action is required for them to perform their function. Internal tank pressure and elevation head are sufficient to discharge the contents to the PCS, if PCS pressure decreases below the SIT pressure.

Each SIT is piped into one PCS cold leg via the injection lines utilized by the High Pressure Safety Injection and Low Pressure Safety Injection (HPSI and LPSI) systems. Each SIT is isolated from the PCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident.

BASES

BACKGROUND (continued)

The SIT gas and water volumes, gas pressure, tank elevation, and outlet pipe size are selected to allow three of the four SITs to partially recover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three SITs are adequate for this function is consistent with the LOCA assumption that the entire contents of one SIT will be lost via the break during the blowdown phase of a LOCA.

APPLICABLE SAFETY ANALYSES

The SITs are credited in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the SITs. Reference to the analyses for these DBAs is used to assess changes to the SITs as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of SI flow. These assumptions include signal generation time, equipment starting times, and delivery time due to system piping. In the early stages of a LOCA with a loss of offsite power, the SITs provide the sole source of makeup water to the PCS. (The assumption of a loss of offsite power is required by regulations.) This is because the LPSI pumps and HPSI pumps cannot deliver flow until the Diesel Generators (DGs) start, come to rated speed, and go through their timed loading sequence. In cold leg breaks, the entire contents of one SIT are assumed to be lost through the break during the blowdown and reflood phases.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the primary coolant pump. During this event, the SITs discharge to the PCS as soon as PCS pressure decreases to below SIT pressure. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA assumes a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated initially by the SITs, with pumped flow then providing continued cooling.

As break size decreases, the SITs and HPSI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the SITs continues to decrease until they are not required, and the HPSI pumps become solely responsible for terminating the temperature increase.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This LCO helps to ensure that the following acceptance criteria, established by 10 CFR 50.46 for the ECCS, will be met following a LOCA:

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;

b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is $\leq$ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. The core is maintained in a coolable geometry.

Since the SITs discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

Since the SITs are passive components, single active failures are not applicable to their operation. The SIT isolation valves, however, are not single failure proof; therefore, whenever the valves are open, power is removed from their operators and the switch is key locked open.

These precautions ensure that the SITs are available during an accident. With power supplied to the valves, a single active failure could result in a valve closure, which would render one SIT unavailable for injection. If the contents of a second SIT is lost through the break, only the contents of two SITs would reach the core. Since the only active failure that could affect the SITs would be the closure of a motor operated outlet valve, the requirement to remove power from these eliminates this failure mode.

The minimum volume requirement for the SITs ensures that three SITs can provide adequate inventory to reflood the core and downcomer following a LOCA. The downcomer then remains flooded until the HPSI and LPSI systems start to deliver flow.

The maximum volume limit is based on maintaining an adequate gas volume to ensure proper injection and the ability of the SITs to fully discharge, as well as limiting the maximum boron inventory in the SITs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The minimum SIT volume of 1040 ft^3 and the maximum SIT volume of 1176 ft^3 correspond to a level of 174 inches and 200 inches, respectively. Each SIT is equipped with two float type level switches which activate control room alarms on high and low level. To allow for instrument inaccuracy, the low SIT level switch alarm is set at 176 inches and the high SIT alarm is set at 198 inches. As a backup to the SIT level switches and to facilitate operator use, level indication is also provided by a differential pressure transmitter which displays in percent tank level. The narrow indicating range of the differential pressure transmitter contains high and low alarms. The high level alarm trips at a slightly lower level than the high level switch and the low level alarm trips at a slightly higher level than the low level switch to alert the operator they are approaching the technical specification values.

The minimum nitrogen cover pressure requirement ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analyses.

A minimum pressure of 200 psig is used in the analyses. Each of the four SITs is equipped with two pressure switches and one pressure transmitter. The pressure switches activate separate control room alarms. One pressure switch provides a high pressure alarm and the other provides a low pressure alarm. The pressure transmitter provides a display of tank pressure and a common high/low pressure alarm. The low pressure alarms from the pressure switch and pressure transmitter are set sufficiently above the 200 psig value used in the safety analysis to provide margin for instrument inaccuracies. The high pressure alarms from the pressure switch and pressure transmitter are set well below the 250 psig tank design pressure and sufficiently above the normal operating pressure to avoid nuisance alarms.

The 1720 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum level in the SITs, the reactor will remain subcritical in the cold condition following mixing of the SITs, Safety Injection Refueling Water Tank and PCS water volumes. Small break LOCAs assume that all full-length control rods are inserted, except for the control rod of highest worth, which is withdrawn from the core. Large break LOCA analyses assume that all full-length control rods remain withdrawn until the blowdown phase is over. For large break LOCAs, the initial reactor shutdown is accomplished by void formation. The most limiting case occurs at beginning of core life.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum boron limit of 2500 ppm in the SITs is based on boron precipitation in the core following a LOCA. With the reactor vessel at saturated conditions, the core dissipates heat by boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point will be reached where boron precipitation will occur in the core. Post LOCA emergency procedures direct the operator to establish simultaneous hot and cold leg injection to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA. Boron concentrations in the SITs in excess of the limit could result in precipitation earlier than assumed in the analysis.

The SITs satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The LCO establishes the minimum conditions required to ensure that the SITs are available to accomplish their core cooling safety function following a LOCA. Four SITs are required to be OPERABLE to ensure that 100% of the contents of three of the SITs will reach the core during a LOCA.

This is consistent with the assumption that the contents of one tank spill through the break. If the contents of fewer than three tanks are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 could be violated.

For an SIT to be considered OPERABLE, the isolation valve must be fully open, with power to the valve operator removed, and the limits established in the SR for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2 the SIT OPERABILITY requirements are based on an assumption of full power operation. Although cooling requirements decrease as power decreases, the SITs are required to be OPERABLE during the MODES when the reactor is critical.

In MODE 3 and below, the rate of PCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 limit of 2200°F.

BASES

APPLICABILITY (continued)

In MODES 3, 4, 5, and 6, the SIT motor operated isolation valves may be closed to isolate the SITs from the PCS. This allows PCS cooldown and depressurization without discharging the SITs into the PCS or requiring depressurization of the SITs.

ACTIONS

A.1

If the boron concentration of one SIT is not within limits, it must be returned to within the limits within 72 hours. In this condition, the ability to maintain subcriticality or minimum boron precipitation time may be reduced, but the reduced concentration effects on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the SIT is still available for injection.

Since the boron requirements are based on the average boron concentration of the total volume of three SITs, the consequences are less severe than they would be if an SIT were not available for injection.

Thus, 72 hours is allowed to return the boron concentration to within limits.

The combination of redundant level and pressure instrumentation for any single SIT provides sufficient information so that it is not worthwhile to always attempt to correct drift associated with one instrument, with the resulting radiation exposures during entry into containment, as there is sufficient time to repair one in the event that a second one became inoperable. Because these instruments do not initiate a safety action, it is reasonable to extend the allowable outage time for them. While technically inoperable, the SIT will be available to fulfill its safety function during this time, and, thus, this Completion Time results in a negligible increase in risk.

B.1

If one SIT is inoperable, for reasons other than boron concentration or the inability to verify level or pressure, the SIT must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three SITs cannot be assumed to reach the core during a LOCA as is assumed in the safety analysis.

CE-NPSD-994 (Ref. 3) provides a series of deterministic and probabilistic findings that support the 24 hour Completion Time as having neglible impact on risk as compared to shorter periods for restoring the SIT to OPERABLE status.

BASES

ACTIONS (continued)

C.1

If the SIT cannot be restored to OPERABLE status 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 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power condition in an orderly manner and without challenging plant systems.

D.1

If more than one SIT is inoperable, the plant is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification every 12 hours that each SIT isolation valve is fully open, as indicated in the control room, ensures that SITs are available for injection and ensures timely discovery if a valve should be partially closed. If an isolation valve is not fully open, the rate of injection to the PCS would be reduced. Although a motor operated valve should not change position with power removed, a closed valve could result in not meeting accident analysis assumptions. A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure the unlikelihood of a mispositioned isolation valve.

SR 3.5.1.2 and SR 3.5.1.3

SIT borated water volume and nitrogen cover pressure should be verified to be within specified limits every 12 hours in order to ensure adequate injection during a LOCA. Due to the static design of the SITs, a 12 hour Frequency usually allows the operator sufficient time to identify changes before the limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

Thirty-one days is reasonable for verification to determine that each SIT's boron concentration is within the required limits, because the static design of the SITs limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.

SR 3.5.1.5

Verification every 31 days that power is removed from each SIT isolation valve operator ensures that an active failure could not result in the undetected closure of an SIT motor operated isolation valve. If this were to occur, only two SITs would be available for injection, given a single failure coincident with a LOCA. Since installation and removal of power to the SIT isolation valve operators is conducted under administrative control, the 31 day Frequency was chosen to provide additional assurance that power is removed.

REFERENCES

1. FSAR, Section 14.17
2. FSAR, Chapter 6.1
3. CE-NPSD-994, "CEOG Joint Applications Report for Safety Injection Tank AOT/STI Extension," May 1995

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of Coolant Accident (LOCA);

b. Control Rod Ejection accident;

c. Loss of secondary coolant accident, including a Main Steam Line Break (MSLB) or Loss of Normal Feedwater; and

d. Steam Generator Tube Rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are two phases of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the Primary Coolant System (PCS) via the cold legs. After the Safety Injection Refueling Water Tank (SIRWT) has been depleted, the recirculation phase is entered as the ECCS suction is automatically transferred to the containment sump.

Two suitably redundant, 100% capacity trains are provided. Each train consists of a High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) subsystem. In MODES 1 and 2, and in MODE 3 with PCS temperature $\geq$ 325°F, both trains must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided in the event of a single active failure.

BASES

BACKGROUND (continued)

Each train of a Safety Injection Signal (SIS) actuates LPSI flow by starting one LPSI pump and opening two LPSI loop injection valves. Each train of an SIS actuates HPSI flow by starting one HPSI pump, opening the four associated HPSI loop injection valves, and closing the pressure control valves associated with each Safety Injection Tank. In addition, each train of a SIS will provide a confirmatory open signal to the normally open Component Cooling Water valves which supply seal and bearing cooling to the LPSI, HPSI, and Containment Spray pumps.

The safety analyses assume that one only train of safety injection is available to mitigate an accident. While operating under the provisions of an ACTION, an additional single failure need not be assumed in assuring that a loss of function has not occurred. Therefore, the LPSI flow assumed in the safety analyses can be met if there is an OPERABLE LPSI flow path from the SIRWT to any two PCS loops. The HPSI flow assumed in the safety analyses can be met if there is an OPERABLE HPSI flow path from the SIRWT to each cold leg. In each case, an OPERABLE flow path must include an OPERABLE pump and an OPERABLE injection valve.

A suction header supplies water from the SIRWT or the containment sump to the ECCS pumps. Separate piping supplies each train. The discharge headers from each HPSI pump divide into four supply lines after entering the containment, one feeding each PCS cold leg. The discharge headers from each LPSI pump combine to supply a common header which divides into four supply lines after entering containment, one feeding each PCS cold leg.

The hot-leg injection piping connects the HPSI Train 1 header and the HPSI Train 2 header to the PCS hot-leg. For long term core cooling after a large LOCA, Hot-leg injection is used to assure that for a large cold-leg PCS break, net core flushing flow can be maintained and excessive boric acid concentration in the core which could result in eventual precipitation and core flow blockage will be prevented. Within a few hours after a LOCA, if shutdown cooling is not in operation, the operator initiates simultaneous hot-leg and cold-leg injection. Hot-leg injection motor-operated valve throttle position and installed flow orifices cause HPSI flows to be split approximately equally between hot- and cold-leg injection paths.

BASES

BACKGROUND (continued)

Motor operated valves are set to maximize the LPSI flow to the PCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the PCS cold legs.

For LOCAs coincident with a loss of off-site power that are too small to initially depressurize the PCS below the shutoff head of the HPSI pumps, the core cooling function is provided by the Steam Generators (SGs) until the PCS pressure decreases below the HPSI pump shutoff head.

During low temperature conditions in the PCS, limitations are placed on the maximum number of HPSI pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

During a large break LOCA, PCS pressure could decrease to < 200 psia in < 20 seconds. The ECCS systems are actuated upon receipt of an SIS. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, all loads will be shed at the time the diesel generators receive an automatic start signal. With load shedding completed, the diesel generator breakers will close automatically when generator voltage approaches a normal operating value. Closing of the breakers will reset the load shedding signals and start the sequencer. The sequencers will initiate operation of the engineered safeguard equipment required for the accident. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive Safety Injection Tanks (SITs) and the Safety Injection Refueling Water Tank (SIRWT), covered in LCO 3.5.1, "Safety Injection Tanks (SITs)," and LCO 3.5.4, "Safety Injection Refueling Water Tank (SIRWT)," provide the cooling water necessary to meet the Palisades Nuclear Plant design criteria (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria, established by 10 CFR 50.46 for ECCSs, will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event.

Both a HPSI and a LPSI subsystem are assumed to be OPERABLE in the large break LOCA analysis at full power (Ref. 2). This analysis establishes a minimum required runout flow for the HPSI and LPSI pumps, as well as the maximum required response time for their actuation. The HPSI pump is also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head requirements at the design point for the HPSI pump. The SGTR and MSLB accident analyses also credit the HPSI pumps, but are not limiting in their design.

The large break LOCA event with a loss of offsite power and a single failure (disabling one ECCS train) establishes the OPERABILITY requirements for the ECCS. During the blowdown stage of a LOCA, the PCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding (during large breaks) or control rod insertion (during small breaks).

Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

On smaller breaks, PCS pressure will stabilize at a value dependent upon break size, heat load, and injection flow. The smaller the break, the higher this equilibrium pressure. In all LOCA analyses, injection flow is not credited until PCS pressure drops below the shutoff head of the HPSI pumps.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO ensures that an ECCS train will deliver sufficient water to match decay heat boiloff rates soon enough to minimize core damage for a large LOCA. It also ensures that the HPSI pump will deliver sufficient water during a small break LOCA and provide sufficient boron to limit the return to power following an MSLB event. For smaller LOCAs, PCS inventory decreases until the PCS can be depressurized below the HPSI pumps' shutoff head. During this period of a small break LOCA, the SGs continue to serve as the heat sink providing core cooling.

ECCS - Operating satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

In MODES 1 and 2, and in MODE 3 with PCS temperature ≥ 325°F, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming there is a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

An ECCS train consists of an HPSI subsystem and a LPSI subsystem. In addition, each train includes the piping, instruments, and controls to ensure the availability of an OPERABLE flow path capable of taking suction from the SIRWT on an SIS and automatically transferring suction to the containment sump upon a Recirculation Actuation Signal (RAS).

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the SIRWT to the PCS, via the HPSI and LPSI pumps and their respective supply headers, to each of the four cold leg injection nozzles is available. During the recirculation phase, a flow path is provided from the containment sump to the PCS via the HPSI pumps. For worst case conditions, the containment building water level alone is not sufficient to assure adequate Net Positive Suction Head (NPSH) for the HPSI pumps. Therefore, to obtain adequate NPSH, a portion of the Containment Spray (CS) pump discharge flow is diverted from downstream of the shutdown cooling heat exchangers to the suction of the HPSI pumps following recirculation during a large break LOCA. In this configuration, the CS pumps and shutdown cooling heat exchangers provide a support function for HPSI flow path OPERABILITY. The OPERABILITY requirements for the CS pumps and shutdown cooling heat exchangers are addressed in LCO 3.6.6, "Containment Cooling Systems." Support system OPERABILITY is addressed by LCO 3.0.6.

The flow path for each train must maintain its designed independence to ensure that no single active failure can disable both ECCS trains.

BASES

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with PCS temperature ≥ 325°F, the ECCS OPERABILITY requirements for the limiting Design Basis Accident (DBA) large break LOCA are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPSI pump performance is based on the small break LOCA, which establishes the pump performance curve and has less dependence on power. The requirements of MODE 2 and MODE 3 with PCS temperature ≥ 325°F, are bounded by the MODE 1 analysis.

The ECCS functional requirements of MODE 3, with PCS temperature < 325°F, and MODE 4 are described in LCO 3.5.3, "ECCS - Shutdown."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "PCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "PCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

Condition A is applicable whenever one LPSI subsystem is inoperable. With one LPSI subsystem inoperable, action must be taken to restore OPERABLE status within 7 days. In this condition, the remaining OPERABLE ECCS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining LPSI subsystem could result in loss of ECCS function. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable LPSI subsystem. While mechanical system LCOs typically provide a 72 hour Completion Time, this 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 5. Reference 5 concluded that extending the Completion Time to 7 days for an inoperable LPSI subsystem provides plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the LPSI subsystem unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

BASES

ACTIONS (continued)

B.1

Condition B is applicable whenever one or more ECCS trains is inoperable for reasons other than one inoperable LPSI subsystem. Action B.1 requires restoration of both ECCS trains, (HPSI and LPSI) to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC study (Ref. 3), assuming that at least 100% of the required ECCS flow (that assumed in the safety analyses) is available. If less than 100% of the required ECCS flow is available, Condition D must also be entered.

Mechanical system LCOs typically provide a 72 hour Completion Time under conditions when a required system can perform its required safety function, but may not be able to do so assuming an additional failure. When operating in accordance with the Required Actions of an LCO Condition, it is not necessary to be able to cope with an additional single failure.

The ECCS can provide one hundred percent of the required ECCS flow following the occurance of any single active failure. Therefore, the ECCS function can be met during conditions when those components which could be deactivated by a single active failure are known to be inoperable. Under that condition, however, the ability to provide the function after the occurance of an additional failure cannot be guaranteed. Therefore, continued operation with one or more trains inoperable is allowed only for a limited time.

C.1 and C.2

Condition C is applicable when the Required Actions of Condition A or B cannot be completed within the required Completion Time. Either Condition A or B is applicable whenever one or more ECCS trains is inoperable. Therefore, when Condition C is applicable, either Condition A or B is also applicable. Being in Conditions A or B, and Condition C concurrently maintains both Completion Time clocks for instances where equipment repair allows exit from Condition C while the plant is still within the applicable conditions of the LCO.

If the inoperable ECCS trains cannot be restored to OPERABLE status within the required Completion Times of Condition A and B, 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 PCS temperature reduce to < 325°F within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

D.1

Condition D is applicable with one or more trains inoperable when there is less than 100% of the required ECCS flow available. Either Condition A or B is applicable whenever one or more ECCS trains is inoperable. Therefore, when this Condition is applicable, either Condition A or B is also applicable. Being in Conditions A or B, and Condition D concurrently maintains both Completion Time clocks for instances where equipment repair allows exit from Condition D (and LCO 3.0.3) while the plant is still within the applicable conditions of the LCO.

One hundred percent of the required ECCS flow can be provided by one OPERABLE HPSI subsystem and one OPERABLE LPSI subsystem. The required LPSI flow (that assumed in the safety analyses) is available if there is an OPERABLE LPSI flow path from the SIRWT to any two PCS loops. Shutdown cooling flow control valve, CV-3006 must be full open. The required HPSI flow (that assumed in the safety analyses) is available if there is an OPERABLE HPSI flow path from the SIRWT to each PCS loop (having less than all four PCS loop flowpaths may be acceptable if verified against current safety analyses). A Containment Spray Pump and a sub-cooled suction valve must be available to support each OPERABLE HPSI pump. In each case, an OPERABLE flow path must include an OPERABLE pump and OPERABLE loop injection valves.

Reference 4 describes situations in which one component, such as the shutdown cooling flow control valve, CV-3006, can disable both ECCS trains. With one or more components inoperable, such that 100% of the required ECCS flow (that assumed in the safety analyses) is not available, the facility is in a condition outside the accident safety analyses.

With less than 100% of the required ECCS flow available, the plant is in a condition outside the assumptions of the safety analyses. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the PCS is maintained. CV-3027 and CV-3056 are stop valves in the minimum recirculation flow path for the ECCS pumps. If either of these valves were closed when the PCS pressure was above the shutoff head of the ECCS pumps, the pumps could be damaged by running with insufficient flow and thus render both ECCS trains inoperable.

Placing HS-3027A and HS-3027B for CV-3027, and HS-3056A and HS-3056B for CV-3056, in the open position ensures that the valves cannot be inadvertently misaligned or change position as the result of an active failure. These valves are of the type described in Reference 4, which can disable the function of both ECCS trains and invalidate the accident analysis. CV-3027 and CV-3056 are capable of being closed from the control room since the SIRWT must be isolated from the containment during the recirculation phase of a LOCA. A 12 hour Frequency is considered reasonable in view of other administrative controls ensuring that a mispositioned valve is an unlikely possibility.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve automatically repositions within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is appropriate because the valves are operated under procedural control and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.5.2.3

SR 3.5.2.3 verifies CV-3006 is in the open position and that its air supply is isolated. CV-3006 is the shutdown cooling flow control valve located in the common LPSI flow path. The valve must be verified in the full open position to support the low pressure injection flow assumptions used in the accident analyses. The inadvertent misposition of this valve could result in a loss of low pressure injection flow and thus invalidate these flow assumptions. CV-3006 is designed to be held open by spring force and closed by air pressure. To ensure the valve cannot be inadvertently misaligned or change position as the result of a hot short in the control circuit, the air supply to CV-3006 is isolated. Isolation of the air supply to CV-3006 is acceptable since the valve does not require automatic repositioning during an accident.

The 31 day Frequency has been shown to be acceptable through operating practice and the unlikely occurrence of the air supply to CV-3006 being unisolated coincident with a inadvertent valve misalignment event or a hot short in the control circuit.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5, SR 3.5.2.6, and SR 3.5.2.7

These SRs demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated actuation signal, i.e., on an SIS or RAS, that each ECCS pump starts on receipt of an actual or simulated actuation signal, i.e., on an SIS, and that the LPSI pumps stop on receipt of an actual or simulated actuation signal, i.e., on an RAS. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.5, SR 3.5.2.6, and SR 3.5.2.7 (continued)

The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability of the equipment and operating experience. The actuation logic is tested as part of the Engineered Safety Feature (ESF) testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.8

The HPSI Hot Leg Injection motor operated valves and the LPSI loop injection valves have position switches which are set at other than the full open position. This surveillance verifies that these position switches are set properly.

The HPSI Hot leg injection valves are manually opened during the post-LOCA long term cooling phase to admit HPSI injection flow to the PCS hot leg. The open position limit switch on each HPSI hot leg isolation valves is set to establish a predetermined flow split between the HPSI injection entering the PCS hot leg and cold legs.

The LPSI loop injection MOVs open automatically on a SIS signal. The open position limit switch on each LPSI loop injection valve is set to establish the maximum possible flow through that valve. The design of these valves is such that excessive turbulence is developed in the valve body when the valve disk is at the full open position. Stopping the valve travel at slightly less than full open reduces the turbulence and results in increased flow. Verifying that the position stops are properly set ensures that a single low pressure safety injection subsystem is capable of delivering the flow rate required in the safety analysis.

The 18 month Frequency is based on the same factors as those stated above for SR 3.5.2.5, SR 3.5.2.6, and SR 3.5.2.7.

SR 3.5.2.9

Periodic inspection of the containment sump ensures that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under outage conditions. This Frequency is sufficient to detect abnormal degradation and is confirmed by operating experience.

BASES

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.17
3. NRC Memorandum to V. Stello, Jr., from R. L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975
4. IE Information Notice No. 87-01, January 6, 1987
5. CE-NPSD-994, "CEOG Joint Applications Report for Safety Injection Tank AOT/STI Extension," May 1995

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

The Background section for Bases B 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

In MODE 3 with Primary Coolant System (PCS) temperature < 325°F and in MODE 4, an ECCS train is defined as one Low Pressure Safety Injection (LPSI) train. The LPSI flow path consists of piping, valves, and pumps that enable water from the Safety Injection Refueling Water Tank (SIRWT), and subsequently the containment sump, to be injected into the PCS following a Loss of Coolant Accident (LOCA).

APPLICABLE SAFETY ANALYSES

In Mode 3 with PCS temperature < 325°F and in Mode 4 the normal compliment of ECCS components is reduced from that which is available during operations above Mode 3 with PCS temperature ≥ 325°F. The acceptability for the reduced ECCS operational requirements is based on engineering judgement rather than specific analysis and considers such factors as the reduced probability that a LOCA will occur, and the reduced energy stored in the fuel. The reduction in ECCS operational requirements include:

1) Isolation of the Safety Injection Tanks (SITs) since PCS pressure is expected to be reduced below the SIT injection pressure,

2) Reliance on manual safety injection initiation since the automatic Safety Injection Signal (SIS) is not required by the technical specifications below 300°F,

3) Rendering the High Pressure Safety Injection (HPSI) pumps incapable of injecting into the PCS. The HPSI pumps are rendered incapable of injecting into the PCS in accordance with the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System". This action assures that a single mass addition event initiated at a pressure within the limits of LCO 3.4.12 cannot cause the PCS pressure to exceed the 10 CFR 50 Appendix G limit.

BASES

APPLICABLE SAFETY ANALYSES (continued)

At a PCS temperature of 325°F the maximum allowed PCS pressure corresponds to the LTOP setpoint limit which is approximately 800 psia. Below 800 psia postulated piping flaws of critical size are considered unlikely since normal operation at 2060 psia serves as a proof test against ruptures. In addition, since the reactor has been shutdown for a period of time, the decay heat and sensible heat levels are greatly reduced from the full power case.

Although a pipe break in the PCS pressure boundary is considered unlikely, break sizes larger and smaller than approximately 0.1 ft^2 are considered separately when analyzing ECCS response.

For breaks larger than approximately 0.1 ft^2, the event is characterized by a very rapid depressurization of the PCS to near the containment pressure. Due to the reduced temperature and pressure of the PCS, the time to complete blowdown is extended from that assumed in the full power case. During this time, the fuel is cooled by the flow through the core towards the break. Automatic safety injection actuation is not assumed to occur since the pressurizer pressure SIS may be bypassed below 1700 psig. Therefore, operator action is relied upon to initiate ECCS flow. Indication that would alert the operator that a LOCA had occurred include; a loss of pressurizer level, rapid decrease in PCS pressure, increase in containment pressure, and containment high radiation alarm. Since the saturation pressure for 325°F is approximately 100 psia, the LPSI pumps are capable of providing the required heat removal function. When the OPERABLE LPSI pump is being used to fulfill the shutdown cooling function, the PCS pressure is < 300 psia. As such, the rate of PCS blowdown is reduced providing some time to manually realign the OPERABLE LPSI pump to the ECCS mode of operation.

For breaks smaller than approximately 0.1 ft^2, the event is characterized by a slow depressurization of the PCS and a relatively long time for the PCS level to drop below the tops of the hot legs. In MODE 3 with PCS temperature < 325°F and in the upper range of MODE 4 before shutdown cooling is established, the spectrum of smaller break sizes are more limiting than larger breaks in terms of ECCS performance since the PCS could stay above the shutoff head of the LPSI pumps. For these break sizes, sufficient time, well in excess of the recommended 10 minutes attributed for manual operator action, is available to either initiate once through cooling using the PORVs, or by re-establishing HPSI pump injection capability. In either case, the core remains covered and the criteria of 10 CFR 50.46 preserved.

ECCS - Shutdown satisfies Criterion 3 of 10 CFR 50.36(c)(2).

BASES

LCO

In MODE 3 with PCS temperature < 325°F and in MODE 4, an ECCS train is comprised of a single LPSI train. Each LPSI train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the SIRWT and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to supply water from the SIRWT to the PCS via one LPSI pump and at least one supply header to a cold leg injection nozzle. In the long term, this flow path may be switched to take its supply from the containment sump.

With PCS temperature < 325°F, one LPSI pump is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements. The High Pressure Safety Injection (HPSI) pumps may therefore be released from the ECCS train requirements. With PCS temperature < 300°F, both HPSI pumps must be rendered incapable of injection into the PCS in accordance with LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The LCO is further modified by a Note that allows a LPSI train to be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation of a LPSI pump in the shutdown cooling mode.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with PCS temperature ≥ 325°F, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 3 with PCS temperature < 325°F and in MODE 4, one OPERABLE ECCS train is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements.

BASES

APPLICABILITY (continued)	In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "PCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "PCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."
ACTIONS	A.1 With no LPSI train OPERABLE, the plant is not prepared to respond to a loss of coolant accident. Action must be initiated Immediately to restore at least one LPSI train to OPERABLE status. The Immediate Completion Time reflects the importance of maintaining an OPERABLE LPSI train and ensures that prompt action is taken to restore the required cooling capacity. B.1 When the Required Action cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.5.3.1 The applicable Surveillance descriptions from Bases 3.5.2 apply.
REFERENCES	The applicable references from Bases 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Safety Injection Refueling Water Tank (SIRWT)

BASES

BACKGROUND

The SIRWT supports the ECCS and the Containment Spray System by providing a source of borated water for Engineered Safety Feature (ESF) pump operation.

The SIRWT supplies two ECCS trains by separate, redundant supply headers. Each header also supplies one train of the Containment Spray System. An air operated isolation valve is provided in each header which isolates the SIRWT from the ECCS after the ESF pump suction has been transferred to the containment sump following depletion of the SIRWT during a Loss of Coolant Accident (LOCA). A separate header is used to supply the Chemical and Volume Control System (CVCS) from the SIRWT. Use of a single SIRWT to supply both trains of the ECCS and Containment Spray System is acceptable since the SIRWT is a passive component, and passive failures are not assumed to occur concurrently with any Design Basis Event during the injection phase of an accident. Not all the water stored in the SIRWT is available for injection following a LOCA; the location of the ESF pump suction piping in the SIRWT will result in some portion of the stored volume being unavailable.

The High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at shutoff head conditions. These lines discharge back to the SIRWT, which vents to the atmosphere. When the suction for the ESF pumps is transferred to the containment sump, the recirculation path must be isolated to prevent is a release of the containment sump contents to the SIRWT. If not isolated, this flow path could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ESF pumps.

BASES

BACKGROUND (continued)

This LCO ensures that:

a. The SIRWT contains sufficient borated water to support ESF pump operation during the injection phase;

b. Sufficient water volume exists in the containment sump to support continued operation of the ESF pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA.

Insufficient water inventory in the SIRWT could result in insufficient cooling capacity of the ECCS when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of shutdown margin or excessive boric acid precipitation in the core following a LOCA, as well as excessive stress corrosion of mechanical components and systems inside containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the SIRWT provides a source of borated water to the HPSI, LPSI, and Containment Spray pumps. As such, it provides containment cooling and depressurization, core cooling, replacement inventory, and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of Bases B 3.5.2, "ECCS - Operating," and B 3.6.6, "Containment Cooling Systems." These analyses are used to assess changes to the SIRWT in order to evaluate their effects in relation to the acceptance limits.

In MODES 1, 2, and 3 the minimum volume limit of 250,000 gallons is based on two factors:

a. Sufficient deliverable volume must be available to provide at least 20 minutes of full flow from one train of ESF pumps prior to reaching a low level switch over to the containment sump for recirculation; and

b. The containment sump water volume must be sufficient to support continued ESF pump operation after the switch over to recirculation occurs. This sump volume water inventory is supplied by the SIRWT borated water inventory.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Twenty minutes is the point at which approximately 75% of the design flow of one HPSI pump is capable of meeting or exceeding the decay heat boiloff rate.

The SIRWT capacity, alone, is not sufficient to provide adequate Net Positive Suction Head (NPSH) for the HPSI pumps after switch over to the containment sump for the worst case conditions. To assure adequate NPSH for the HPSI pumps, their suction headers are aligned to the discharge of the Containment Spray Pumps (Ref. 2). Restrictions are placed on Containment Spray Pump operation with this alignment to ensure the Containment Spray Pumps have adequate NPSH (Ref. 3).

In MODE 4, the minimum volume limit of 200,000 gallons is based on engineering judgement and considers factors such as:

a. The volume of water transferred from the SIRWT to the PCS to account for the change in PCS water volume during a cooldown from 532°F to 200°F (approximately 17,000 gallons assuming an initial PCS volume of 80,000 gallons); and

b. The minimum SIRWT water volume capable of providing a sufficient level in the containment sump to support LPSI pump operation following a LOCA.

Due to the reduced PCS temperature and pressure requirements in MODE 4, and in recognition that water from the SIRWT used for PCS makeup is available for recirculation following a LOCA, the minimum water volume limit for the SIRWT in MODE 4 is lower than in MODES 1, 2, or 3.

The 1720 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum level in the SIRWT, the reactor will remain subcritical in the cold condition following mixing of the SIRWT, Safety Injection Tanks, and PCS water volumes. Small break LOCAs assume that all full-length control rods are inserted, except for the control rod of highest worth, which is withdrawn from the core. Large break LOCA analyses assume that all full-length control rods remain withdrawn until the blowdown phase is over. For large break LOCAs, the initial reactor shutdown is accomplished by void formation. The most limiting case occurs at beginning of core life.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum boron limit of 2500 ppm in the SIRWT is based on boron precipitation in the core following a LOCA. With the reactor vessel at saturated conditions, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point will be reached where boron precipitation will occur in the core. Post LOCA emergency procedures direct the operator to establish simultaneous hot and cold leg injection to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA. Boron concentrations in the SIRWT in excess of the limit could result in precipitation earlier than assumed in the analysis.

SIRWT boron concentration and volume also determine the post-LOCA pump pH. Trisodium Phosphate (TSP), stored in the lower region of containment, mixes with the SIRWT water following a LOCA to control pH. Maintaining pH in the proper range is necessary to retain iodine in solution, prevent excessive hydrogen generation, and to prevent potential long term stress corrosion cracking in ESF piping. TSP requirements are addressed in LCO 3.5.5, "Trisodium Phosphate (TSP).".

The upper limit of 100°F and the lower limit of 40°F on SIRWT temperature are the limits assumed in the accident analysis. SIRWT temperature affects the outcome of several analyses. Although the minimum temperature limit of 40°F was selected to maintain a small margin above freezing (32°F), violation of the minimum temperature could result in unacceptable conclusions for some analyses. The upper temperature limit of 100°F is used in the Containment Pressure and Temperature Analysis. Exceeding this temperature will result in higher containment pressure due to reduced containment spray cooling capacity.

The SIRWT satisfies Criterion 3 of 10 CFR 50.36(c)(2).

BASES

LCO

The SIRWT ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, that the reactor remains subcritical following a DBA, and that an adequate level exists in the containment sump to support ESF pump operation in the recirculation mode.

To be considered OPERABLE, the SIRWT must meet the limits established in the SRs for water volume, boron concentration, and temperature.

APPLICABILITY

In MODES 1, 2, and 3, the SIRWT OPERABILITY requirements are dictated by the ECCS and Containment Spray System OPERABILITY requirements. In MODE 4 the SIRWT OPERABILITY requirements are dictated by ECCS requirements only. As such, the SIRWT must be OPERABLE in MODES 1, 2, 3, and 4.

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "PCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "PCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With SIRWT boron concentration or borated water temperature not within limits, it must be returned to within limits within 8 hours. In this condition neither the ECCS nor the Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE condition. The allowed Completion Time of 8 hours to restore the SIRWT to within limits was developed considering the time required to change boron concentration or temperature, and that the contents of the tank are still available for injection.

BASES

ACTIONS (continued)

B.1

With SIRWT borated water volume not within limits, it must be returned to within limits within 1 hour. In this condition, neither the ECCS nor Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which these systems are not required. The allowed Completion Time of 1 hour to restore the SIRWT to OPERABLE status is based on this condition simultaneously affecting multiple redundant trains.

C.1 and C.2

If the SIRWT cannot be restored to OPERABLE status 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 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.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

SIRWT borated water temperature shall be verified every 24 hours to be within the limits assumed in the accident analysis. This Frequency has been shown to be sufficient to identify temperature changes that approach either acceptable limit.

SR 3.5.4.2 and SR 3.5.4.3

The minimum SIRWT water volume shall be verified every 7 days. This Frequency ensures that a sufficient initial water supply is available for injection and to support continued ESF pump operation on recirculation. Since the SIRWT volume is normally stable and is provided with a Low Level Alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.2 is modified by a Note which states that it is only required to be met in MODES 1, 2, and 3.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.4.2 and SR 3.5.4.3 (continued)

SR 3.5.4.3 is modified by a Note which states that it is only required to be met in MODE 4. The required minimum SIRWT water volume is less in MODE 4 since the PCS temperature and pressure are reduced and a significant volume of water is transferred from the SIRWT to the PCS during MODE 4 to account for primary coolant shrinkage.

SR 3.5.4.4

Boron concentration of the SIRWT shall be verified every 31 days to be within the required range. This Frequency ensures that the reactor will remain subcritical following a LOCA. Further, it ensures that the resulting sump pH will be maintained in an acceptable range such that boron precipitation in the core will not occur earlier than predicted and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized.

Since the SIRWT volume is normally stable, a 31 day sampling Frequency is appropriate and has been shown through operating experience to be acceptable.

REFERENCES

1. FSAR, Chapter 6 and Chapter 14
2. Design Basis Document (DBD) 2.02, "High-Pressure Safety Injection System," Section 3.3.1
3. EOP 4.0, Loss of Coolant Accident

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Trisodium Phosphate (TSP)

BASES

BACKGROUND

TSP baskets are is placed on the floor (590 ft elevation) in the containment building to ensure that iodine, which may be dissolved in the recirculated primary cooling water following a Loss of Coolant Accident (LOCA), remains in solution (Ref. 1). Recirculation of the water for core cooling and containment spray provides mixing to achieve a uniform neutral pH. TSP also helps inhibit Stress Corrosion Cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident.

Fuel that is damaged during a LOCA will release iodine in several chemical forms to the reactor coolant and to the containment atmosphere. A portion of the iodine in the containment atmosphere is washed to the sump by containment sprays. The Safety Injection Refueling Water Tank water is borated for reactivity control. This borated water, if left untreated, would cause the sump solution to be acidic. In a low pH (acidic) solution, dissolved iodine will be converted to a volatile form. The volatile iodine will evolve out of solution into the containment atmosphere, significantly increasing the levels of airborne iodine. The increased levels of airborne iodine in containment contribute to the radiological releases and increase the consequences from the accident due to containment atmosphere leakage.

After a LOCA, the components of the safety injection and containment spray systems will be exposed to high temperature borated water. Prolonged exposure to hot untreated sump water combined with stresses imposed on the components can cause SCC. The rate of SCC is a function of stress, oxygen and chloride concentrations, pH, temperature, and alloy composition of the components. High temperatures and low pH, which would be present after a LOCA, tend to promote SCC. This can lead to the failure of necessary safety systems or components.

BASES

BACKGROUND (continued)	Adjusting the pH of the recirculation solution to levels above 7.0 prevents a significant fraction of the dissolved iodine from converting to a volatile form. The higher pH thus decreases the level of airborne iodine in containment and reduces the radiological consequences from containment atmosphere leakage following a LOCA. Maintaining the solution pH above 7.0 also reduces the occurrence of SCC of austenitic stainless steel components in containment. Reducing SCC reduces the probability of failure of components. The hydrated form (45-57% moisture) of TSP is used because of the high humidity in the containment building during normal operation. Since the TSP is hydrated, it is less likely to absorb large amounts of water from the humid atmosphere and will undergo less physical and chemical change than the anhydrous form of TSP.
APPLICABLE SAFETY ANALYSES	The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being ≥ 7.0. The radionuclide releases from the containment atmosphere and the consequences of a LOCA would be increased if the pH of the recirculation water were not adjusted to 7.0 or above. The containment hydrogen concentration analysis used in the evaluation of the Maximum Hypothetical Accident (MHA) assumes the pH of the containment sump water is between 7.0 and 8.0. The acceptance criteria of the MHA includes a containment lower flammability limit of 4 volume percent for hydrogen. Containment sump water with a pH greater than 8.0 could result in excess hydrogen generation in containment and invalidate the conclusions of the MHA. TSP satisfies Criterion 3 of 10 CFR 50.36(c)(2).
LCO	The quantity of TSP placed in containment is designed to adjust the pH of the sump water to be between 7.0 and 8.0 after a LOCA. A pH > 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH > 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

BASES

LCO (continued)

The pH needs to remain < 8.0 to remain within the assumptions of the analysis for post-LOCA Hydrogen concentration in the containment.

The minimum acceptable amount of TSP is that weight which will ensure a sump solution pH ≥ 7.0 after a LOCA, with the maximum amount of water at the minimum initial pH possible in the containment sump; a maximum acceptable amount of TSP is that weight which will ensure a sump solution pH of ≤ 8.0 with a minimum amount of water at a maximum initial pH.

The TSP is stored in wire mesh baskets placed inside the containment at the 590 ft elevation. Any quantity of TSP between 8,300 and 11,000 lb. will result in a pH in the desired range.

APPLICABILITY

In MODES 1, 2, and 3, the PCS is at elevated temperature and pressure, providing an energy potential for a LOCA. The potential for a LOCA results in a need for the ability to control the pH of the recirculated coolant.

In MODES 4, 5, and 6, the potential for a LOCA is reduced or nonexistent, and TSP is not required.

ACTIONS

A.1

If it is discovered that the TSP in the containment building is not within limits, action must be taken to restore the TSP to within limits.

The Completion Time of 72 hours is allowed for restoring the TSP within limits, where possible, because 72 hours is the same time allowed for restoration of other ECCS components.

B.1 and B.2

If the TSP cannot be restored within limits within the Completion Time of Required Action A.1, the plant must be brought to a MODE in which the LCO does not apply. The specified Completion Times for reaching MODES 3 and 4 are those used throughout the Technical Specifications; they were chosen to allow reaching the specified conditions from full power in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Periodic determination of the mass of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. A Frequency of 18 months is required to determine that ≥ 8,300 lbs and ≤ 11,000 lbs are contained in the TSP baskets. This requirement ensures that there is an adequate mass of TSP to adjust the pH of the post LOCA sump solution to a value ≥ 7.0 and ≤ 8.0.

The periodic verification is required every 18 months, since determining the mass of the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 18 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the mass of TSP placed in the containment building.

SR 3.5.5.2

Periodic testing is performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. Satisfactory completion of this test assures that the TSP in the baskets is "active."

Adequate solubility is verified by submerging a representative sample of TSP from one of the baskets in containment in un-agitated borated water heated to a temperature representing post-LOCA conditions; the TSP must completely dissolve within a 4 hour period. The test time of 4 hours is to allow time for the dissolved TSP to naturally diffuse through the un-agitated test solution. Agitation of the test solution during the solubility verification is prohibited, since an adequate standard for the agitation intensity (other than no agitation) cannot be specified. The agitation due to flow and turbulence in the containment sump during recirculation would significantly decrease the time required for the TSP to dissolve.

Adequate buffering capability is verified by a measured pH of the sample solution, following the solubility verification, between 7.0 and 8.0. The sample is cooled and thoroughly mixed prior to measuring pH.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.5.2 (continued)

The quantity of the TSP sample, and quantity and boron concentration of the water are chosen to be representative of post-LOCA conditions.

A sampling Frequency of every 18 months is specified. Operating experience has shown this Surveillance Frequency to be acceptable.

REFERENCES

1. FSAR, Section 6.4

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of a concrete structure lined with steel plate, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis Loss of Coolant Accident (LOCA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The foundation slab is reinforced with conventional mild-steel reinforcing. The internal pressure loads on the base slab are resisted by both the external soil pressure and the strength of the reinforced concrete slab. The cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions. The dome roof is prestressed utilizing a three way post tensioning system. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The concrete structure is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 4) as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

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

    1. capable of being closed by an OPERABLE automatic containment isolation system, or

    2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

BASES

BACKGROUND (continued)	b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; c. The equipment hatch is properly closed and sealed.
APPLICABLE SAFETY ANALYSES	The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate. The DBAs that result in a release of radioactive material within containment are a Loss of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), and a control rod ejection accident (Ref. 1). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.10% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B as L_a; the maximum allowable leakage rate at pressure P_a. The P_a value of 53 psig represents the analytical value found in Reference 1, rounded up to the next whole number. Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY. The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2).
LCO	Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Containment Leak Rate Testing Program leakage test. At this time, the applicable leakage limits must be met. Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Individual leakage rates specified for the containment air lock (LCO 3.6.2) and purge valves which have resilient seals (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of $1.0\ L_a$.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABILITY during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring, during periods when containment is inoperable, is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leak Rate Testing Program. Failure to meet individual air lock and containment isolation valve "local leak rate" leakage limits does not invalidate the acceptability of the overall leakage determination unless their contribution to overall Type A, B, or C leakage causes that leakage to exceed limits. As left leakage prior to the first startup after performing a required Containment Leak Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leak Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Structural Integrity Surveillance Program.

REFERENCES

1. FSAR, Chapter 14
2. FSAR, Section 14.18
3. FSAR, Section 5.8
4. 10 CFR 50, Appendix J, Option B

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Two air locks provide access into the containment. Each air lock is nominally a right circular cylinder, with a door at each end. The personnel air lock doors are 3 foot, 6 inches by 6 foot, 8 inches. The emergency escape air lock doors are 30 inches in diameter. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the plant safety analysis.

BASES

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a Loss of Coolant Accident (LOCA), a Main Steam Line Break (MSLB) and a control rod ejection accident (Ref. 1). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.10% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B, as L_a : the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a). For a LOCA, the calculated maximum peak containment pressure is approximately 53 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single OPERABLE door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by three notes. The first note allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, even if this door has been locked to comply with ACTIONS. This means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable because of the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions. A third Note has been included that requires entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage limit.

BASES

ACTIONS (continued)

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

With one air lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed an OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per. . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.6.2 A.3 must be initially performed within 31 days without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception provided by Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions.

Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

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

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. If the overall containment leakage rate exceeds the limits of LCO 3.6.1, the conditions of that LCO must be entered in accordance with Actions Note 3. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

BASES

ACTIONS (continued)

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leak Rate Testing Program.

This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria, were established during initial air lock and containment Operability testing. Subsequent amendments to the Technical Specifications revised the acceptance criteria for overall Type B and C leakage limits and provided new acceptance criteria for the personnel air lock doors and the emergency air lock doors (Ref. 2). The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leak Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit into and out of containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the airlock is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month frequency is based on the need to perform this Surveillance under the conditions that apply during plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power.

The 24 month Frequency for the interlock is justified based on generic operating experience. The Frequency is based on engineering judgment and is considered adequate given that the interlock is not normally challenged during use of the airlock.

REFERENCES

1. FSAR, Chapter 14
2. FSAR, Section 5.8
3. 10 CFR 50, Appendix J, Option B

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves and devices form part of the containment pressure boundary and provide a means for isolating penetration flow paths. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. One of these barriers may be a closed system.

Containment isolation occurs upon receipt of a Containment High Pressure (CHP) signal or a Containment High Radiation (CHR) signal. However, not all containment isolation valves are actuated by both signals. The signals close automatic containment isolation valves in fluid penetrations not required for operation of Engineered Safety Feature systems in order to prevent leakage of radioactive material. Other penetrations are isolated by the use of valves or check valves in the closed position, or blind flanges. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated in the event of a release of radioactive material to containment atmosphere from the Primary Coolant System (PCS) following a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves and devices help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that the containment leakage limits assumed in the accident analysis will be not exceeded in a DBA.

BASES

BACKGROUND (continued)	The 8 inch purge exhaust valves are designed for purging the containment atmosphere to the stack while introducing filtered makeup, through the 12 inch air room supply valves from the outside, when the plant is shut down during refueling operations and maintenance. The purge exhaust valves and air room supply valves are air operated isolation valves located outside the containment. These valves are operated manually from the control room. These valves will close automatically upon receipt of a CHP or CHR signal. The air operated valves fail closed upon a loss of air. These valves are not qualified for automatic closure from their open position under DBA conditions. Therefore, these valves are locked closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained. Open purge exhaust or air room supply valves, following an accident that releases contamination to the containment atmosphere, would cause a significant increase in the containment leakage rate.
APPLICABLE SAFETY ANALYSES	The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO. The DBAs that result in a release of radioactive material within containment are a Loss of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), and a control rod ejection accident. In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analysis assumes that the purge exhaust and air room supply valves are closed at event initiation. The DBA analysis assumes that, within 25 seconds after receiving a CHP or CHR signal each automatic power operated valve is closed and containment leakage terminated except for the design leakage rate, L_a.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the design of the containment purge valves. Two valves in series on each line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. Both isolation valves on the 8 inch and 12 inch lines are pneumatically operated spring closed valves.

The 8 inch purge exhaust and 12 inch air room supply valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain locked closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valves due to the potential for failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve safety function is related to minimizing the loss of primary coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate upon receipt of a CHP or CHR signal as appropriate. The purge exhaust and air room supply valves must be locked closed. The valves covered by this LCO are listed with their associated stroke times in the FSAR (Ref. 1).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves or devices are those listed in Reference 1.

The purge exhaust and air room supply valves with resilient seals must meet the same leakage rate testing requirements as other Type C tested containment isolation valves addressed by LCO 3.6.1, "Containment."

BASES

LCO (continued)	This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of primary coolant inventory and establish the containment boundary during accidents.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."
ACTIONS	The ACTIONS are modified by four notes. Note 1 allows isolated penetration flow paths, except for 8 inch exhaust and 12 inch air room supply purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the fact that the 8 inch purge exhaust valves and the 12 inch air room supply valves may be unable to close in the environment following a LOCA and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, these valves may not be opened under administrative controls. Note 2 to provides clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions. Note 3,ensures that appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve. Note 4 requires entry into the applicable Conditions and Required Actions of LCO 3.6.1 when leakage results in exceeding the overall containment leakage limit.

BASES

ACTIONS (continued)

A.1 and A.2

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides appropriate actions.

In the event one containment isolation valve in one or more penetration flow paths is inoperable (except for purge exhaust or air room supply valves), the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low.

For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

The Completion Time of once per 31 days for verifying each affected penetration flow path outside the containment is isolated is appropriate considering that the penetration can be isolated by the remaining isolable valve. As stated in SR 3.02, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per. . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, for devices outside the containment, while Required Action 3.6.3 A.2 must be initially performed within 31 days without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

B.1

With two containment isolation valves in one or more penetration flow paths inoperable (except for purge exhaust valve or air room supply valve not locked closed), the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.

In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated.

The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

BASES

ACTIONS

B.1 (continued)

Condition B is modified by a Note indicating this Condition is only penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference 2. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated.

The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the valves are operated under administrative controls and the probability of their misalignment is low. As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per. . ." however, the 25% extension does not apply to the initial performance on a "once per. . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.6.3 C.2 must be initially performed within 31 days without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

BASES

ACTIONS

C.1 and C.2 (continued)

Required Action C.2 is modified by a Note that applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1

The purge exhaust and air room supply isolation valves have not been qualified to close following a LOCA and are required to be locked closed. If one or more of these valves is found not locked closed, the potential exists for the valves to be inadvertently opened. One hour is provided to lock closed the affected valves. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining these valves closed.

E.1 and E.2

If the Required Actions and associated Completion Times are not met, 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

This SR ensures that the 8 inch purge exhaust and 12 inch air room supply valves are locked closed as required. If a valve is open, or closed but not locked, in violation of this SR, the valve is considered inoperable. Valves may be locked closed electrically, mechanically, or by other physical means. These valves may be unable to close in the environment following a LOCA. Therefore, each of the valves is required to remain closed during MODES 1, 2, 3, and 4. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.2.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2

This SR requires verification that each manual containment isolation valve and blind flange located outside containment, and not locked, sealed, or otherwise secured in position, and required to be closed during accident conditions, is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed or otherwise secured in position, and required to be closed during accident conditions, is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time that they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3 (continued)

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.5

For containment 8 inch purge exhaust and 12 inch air room supply valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B (Ref. 3), is required to ensure the valves are physically closed (SR 3.6.3.1 verifies the valves are locked closed). Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 4) as specified in the Safety Evaluation for Amendment No. 90 to the Facility Operating License.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures each automatic containment isolation valve will actuate to its isolation position on an actual or simulated actuation signal, i.e., CHP or CHR. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency was developed considering it is prudent that this SR be performed only during a plant outage, since isolation of penetrations would eliminate cooling water flow and disrupt normal operation of many critical components. Operating experience has shown that these components usually pass this SR when performed on the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 5.8
2. FSAR, Section 6.7.2
3. 10 CFR 50, Appendix J, Option B
4. Generic Issue B-20

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB).

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered for determining the maximum containment internal pressure are the LOCA and MSLB. An MSLB at 0% RTP with the MSIVs open results in the highest calculated internal containment pressure of 54.3 psig, which is below the internal design pressure of 55 psig. The postulated DBAs are analyzed assuming degraded containment Engineered Safety Feature (ESF) systems (i.e., assuming the limiting single active failure). While the maximum containment internal pressure results from an MSLB, the licensing basis dose limitations are based on the LOCA (see the Bases for 3.6.1, "Containment," for a discussion on containment pressures resulting from a LOCA).

The initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig) in MODES 1 and 2 and 16.2 psia (1.5 psig in MODES 3 and 4). The LCO limits of 1.0 psig in MODES 1 and 2, and 1.5 psig in MODES 3 and 4 ensures that, in the event of an accident, the maximum accident design pressure for containment, 55 psig, is not exceeded.

A higher containment pressure limit is allowed in MODES 3 and 4 where the reactor is not critical and the resulting heat addition to containment in a DBA is lower.

APPLICABLE SAFETY ANALYSES (continued)

The external design pressure of the containment shell is 3 psig. This value is approximately 0.5 psig greater than the maximum external pressure that could be developed if the containment were sealed during a period of low barometric pressure and high temperature and, subsequently, the containment atmosphere were cooled with a concurrent major rise in barometric pressure. Vacuum breakers are, therefore, not provided and no minimum containment pressure specification is required.

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

Maintaining containment pressure less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Two limits for containment pressure are provided to reflect the analyses which allow for a higher containment pressure when the reactor is not critical due to less heat input to containment in the event of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analysis are maintained, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

BASES

ACTIONS (continued)

B.1 and B.2

If containment pressure cannot be restored to within limits 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that operation remains within the limits assumed in the accident analyses. The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition. The limit of 1.0 psig for MODES 1 and 2, 1.5 psig for MODES 3 and 4 are the actual limits used in the accident analysis and do not account for instrument inaccuracies.

REFERENCES

1. FSAR, Section 14.18

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB).

Containment air temperature is a process variable that is monitored and controlled. The containment average air temperature limit is derived from the input conditions used in the containment accident analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during plant operations. The total amount of energy to be removed from containment by the Containment Spray and Cooling systems during post accident conditions is dependent on the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in a higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis (Ref. 1). Operation with containment average air temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analysis for containment. The accident analyses and evaluations considered both LOCAs and MSLBs for determining the maximum peak containment pressures and temperatures. The worst case MSLB generates larger mass and energy releases than the worst case LOCA. Thus, the MSLB event bounds the LOCA event from the containment peak pressure and temperature standpoint.

The initial pre-accident temperature inside containment was assumed to be 140°F (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The initial containment average air temperature condition of 140°F resulted in a maximum vapor temperature in containment of 401°F. This value represents the analytical value presented in Reference 3, rounded up to the next highest number. This exceeds the containment building design temperature of 283°F. The effect on the containment structure is negligible due to the short period of time the temperature exceeds the design value.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident pressure is maintained below the containment design pressure. As a result, the ability of containment to perform its function is ensured.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

BASES

ACTIONS (continued)

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. The 140°F limit is the actual limit assumed for the accident analyses and does not account for instrument inaccuracies. Instrument uncertainties are accounted for in the surveillance procedure. The 24 hour Frequency of this SR is considered acceptable based on the observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment).

REFERENCES

1. FSAR, Section 5.8
2. FSAR, Section 14.18
3. FSAR, Table 14.18.2-3

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Air Cooler systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a Main Steam Line Break (MSLB) or a large break Loss of Coolant Accident (LOCA). The Containment Spray and Containment Air Cooler systems are designed to the requirements of the Palisades Nuclear Plant design criteria (Ref. 1).

The Containment Air Cooler System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The systems are arranged with two spray pumps and one air cooler fan powered from one diesel generator, and with one spray pump and three air cooler fans powered from the other diesel generator. The Containment Spray System was originally designed to be redundant to the Containment Air Coolers (CACs) and fans. These systems were originally designed such that either two containment spray pumps or three CACs could limit containment pressure to less than design. However, the current safety analyses take credit for one containment spray pump when evaluating cases with three CACs, and for one air cooler fan in cases with two spray pumps and both Main Steam Isolation Valve (MSIV) bypass valves closed. If an MSIV bypass valve is open, 2 service water pumps and 2 CACs are also required to be OPERABLE in addition to the 2 spray pumps for containment heat removal.

To address this dependency between the Containment Spray System and the Containment Air Cooler System the title of this Specification is "Containment Cooling Systems," and includes both systems. The LCO is written in terms of trains of containment cooling. One train of containment cooling is associated with Diesel Generator 1-1 and includes Containment Spray Pumps P-54B and P-54C, Containment Spray Valve CV-3001 and the associated spray header, and Air Cooler Fan V-4A. The other train of containment cooling is associated with Diesel Generator 1-2 and includes Containment Spray Pump P-54A, Containment Spray Valve CV-3002 and the associated spray header, and CACs VHX-1, VHX-2, and VHX-3 and their associated safety related fans, V-1A, V-2A, and V-3A.

BASES

BACKGROUND (continued)

If reliance is placed solely on one spray pump and three CACs, at least two service water pumps must be OPERABLE to provide the necessary service water flow to assure OPERABILITY of the CACs. Additional details of the required equipment and its operation is discussed with the containment cooling system with which it is associated.

Containment Spray System

The Containment Spray System consists of three half-capacity (50%) motor driven pumps, two shutdown cooling heat exchangers, two spray headers, two full sets of full capacity (100%) nozzles, valves, and piping, two full capacity (100%) pump suction lines from the Safety Injection and Refueling Water Tank (SIRWT) and the containment sump with the associated piping, valves, power sources, instruments, and controls. The heat exchangers are shared with the Shutdown Cooling System. SIRWT supplies borated water to the containment spray during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the SIRWT to the containment sump.

Normally, both Shutdown Cooling Heat Exchangers must be available to provide cooling of the containment spray flow in the event of a Loss of Coolant Accident. If the Containment Spray side (tube side) of one SDC Heat Exchanger is out of service, 100% of the required post accident cooling capability can be provided, if other equipment outages are limited. One hundred percent of the post accident cooling can be provided with the Containment Spray side of one SDC Heat Exchanger out of service if the following equipment is OPERABLE: 3 safety related Containment Air Coolers, 2 Containment Spray Pumps, 3 CCW pumps, 2 SWS pumps, and both CCW Heat Exchangers, and if

1. One CCW Containment Isolation Valve, CV-0910, CV-0911, or CV-0940, is OPERABLE, and

2. One CCW header isolation valve for the non-safety related loads outside the containment, CV-0944, CV-0944A, or CV-0977B, is OPERABLE.

The Containment Spray System provides a spray of cold borated water into the upper regions of containment to reduce the containment pressure and temperature during a MSLB or large break LOCA event. In addition, the Containment Spray System in conjunction with the use of trisodium phosphate (LCO 3.5.5, "Trisodium Phosphate,") serve to remove iodine which may be released following an accident. The SIRWT solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase.

BASES

BACKGROUND

In the recirculation mode of operation, heat is removed from the containment sump water by the shutdown cooling heat exchangers.

The Containment Spray System is actuated either automatically by a Containment High Pressure (CHP) signal or manually. An automatic actuation opens the containment spray header isolation valves, starts the three containment spray pumps, and begins the injection phase. Individual component controls may be used to manually initiate Containment Spray. The injection phase continues until an SIRWT Level Low signal is received. The Low Level signal for the SIRWT generates a Recirculation Actuation Signal (RAS) that aligns valves from the containment spray pump suction to the containment sump. The Containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

The containment spray pumps also provide a required support function for the High Pressure Safety Injection pumps as described in the Bases for specification 3.5.2. The High Pressure Safety Injection pumps alone may not have adequate NPSH after a postulated accident and the realignment of their suctions from the SIRWT to the containment sump. Provision is made to manually provide flow from the discharge of the containment spray pumps to the suction of the High Pressure Safety Injection pumps after the change to recirculation mode has occurred. The additional suction pressure ensures that adequate NPSH is available for the High Pressure Safety Injection pumps.

<u>Containment Air Cooler System</u>

The Containment Air Cooler System includes four air handling and cooling units, referred to as the Containment Air Coolers (CACs), which are located entirely within the containment building. Three of the CACs (VHX-1, VHX-2, and VHX-3) are safety related coolers and are cooled by the critical service water. The fourth CAC (VHX-4) is not taken credit for in maintaining containment temperature within limit (the service water inlet valve for VHX-4 is closed by an SIS signal to conserve service water flow), but is used during normal operation along with the three CACs to maintain containment temperature below the design limits. The fan associated with VHX-4, V-4A, is assumed in the safety analysis as assisting in the containment atmosphere mixing function.

The DG which powers the fans associated with VHX-1, VHX-2, and VHX-3 (V-1A, V-2A and V-3A) also powers two service water pumps. This is necessary because if reliance is placed solely on the train with

BACKGROUND Containment Air Cooler System (continued)

one spray pump and three CACs, at least two service water pumps must be OPERABLE to provide the necessary service water flow to assure OPERABILITY of the CACs.

Each CAC has two vaneaxial fans with direct connected motors which draw air through the cooling coils. Both of these fans are normally in operation, but only one fan and motor for each CAC is rated for post accident conditions. The post accident rated "safety related" fan units, V-1A, V-2A, V-3A, and V-4A, serve not only to provide forced flow for the associated cooler, but also provide mixing of the containment atmosphere. A single operating safety related fan unit will provide enough air flow to assure that there is adequate mixing of unsprayed containment areas to assure the assumed iodine removal by the containment spray. The fan units also support the functioning of the hydrogen recombiners, as discussed in the Bases for LCO 3.6.7, "Hydrogen Recombiners." In post accident operation following a SIS, all four Containment air coolers are designed to change automatically to the emergency mode.

The CACs are automatically changed to the emergency mode by a Safety Injection Signal (SIS). This signal will trip the normal rated fan motor in each unit, open the high-capacity service water discharge valve from VHX-1, VHX-2, and VHX-3, and close the high-capacity service water supply valve to VHX-4. The test to verify the service water valves actuate to their correct position upon receipt of an SIS signal is included in the surveillance test performed as part of Specification 3.7.8, "Service Water System." The safety related fans are normally in operation and only receive an actuation signal through the DBA sequencers following an SIS in conjunction with a loss of offsite power. This actuation is tested by the surveillance which verifies the energizing of loads from the DBA sequencers in Specification 3.8.1, "AC Sources-Operating."

BASES

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Air Cooler System limit the temperature and pressure that could be experienced following either a Loss of Coolant Accident (LOCA) or a Main Steam Line Break (MSLB). The large break LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients.

The Containment Cooling Systems have been analyzed for three accident cases (Ref. 2). All accidents analyses account for the most limiting single active failure.

1. A Large Break LOCA,

2. An MSLB occurring at various power levels with both MSIV bypass valves closed, and

3. An MSLB occurring at 0% RTP with both MSIV bypass valves open.

The postulated large break LOCA is analyzed, in regard to containment ESF systems, assuming the loss of offsite power and the loss of one ESF bus, which is the worst case single active failure, resulting in one train of Containment Cooling being rendered inoperable (Ref. 6).

The postulated MSLB is analyzed, in regard to containment ESF systems, assuming the worst case single active failure .

The MSLB event is analyzed at various power levels with both MSIV bypass valves closed, and at 0% RTP with both MSIV bypass valves open. Having any MSIV bypass valve open allows additional blowdown from the intact steam generator.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure and the peak containment vapor temperature are within the intent of the design basis. (See the Bases for Specifications 3.6.4, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations considered a range of power levels and equipment configurations as described in Reference 2. The peak containment pressure case is the 0% power MSLB with initial (pre-accident) conditions of 140°F and 16.2 psia. The peak temperature case is the 102% power MSLB with initial (pre-accident) conditions of 140°F and 15.7 psia. The analyses also assume a response time delayed initiation in order to provide conservative peak calculated containment pressure and temperature responses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The external design pressure of the containment shell is 3 psig. This value is approximately 0.5 psig greater than the maximum external pressure that could be developed if the containment were sealed during a period of low barometric pressure and high temperature and, subsequently, the containment atmosphere was cooled with a concurrent major rise in barometric pressure.

The modeled Containment Cooling System actuation from the containment analysis is based on a response time associated with exceeding the Containment High Pressure setpoint to achieve full flow through the CACs and containment spray nozzles. The spray lines within containment are maintained filled to the 735 ft elevation to provide for rapid spray initiation. The Containment Cooling System total response time of < 60 seconds includes diesel generator startup (for loss of offsite power), loading of equipment, CAC and containment spray pump startup, and spray line filling.

The performance of the Containment Spray System for post accident conditions is given in Reference 3. The performance of the Containment Air Coolers is given in Reference 4.

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

During an MSLB or large break LOCA event, a minimum of one containment cooling train is required to maintain the containment peak pressure and temperature below the design limits (Ref. 2). One train of containment cooling is associated with Diesel Generator 1-1 and includes Containment Spray Pumps P-54B and P-54C, Containment Spray Valve CV-3001 and the associated spray header, and air cooler fan V-4A. This train must be supplemented with 2 service water pumps and 2 containment air coolers if an MSIV bypass valve is open. The other train of containment cooling is associated with Diesel Generator 1-2 and includes Containment Spray Pump P-54A, Containment Spray Valve CV-3002 and the associated spray header, and CACs VHX-1, VHX-2, and VHX-3 and their associated safety related fans, V-1A, V-2A, and V-3A. To ensure that these requirements are met, two trains of containment cooling must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst case single active failure occurs.

BASES

LCO (continued)

The Containment Spray System portion of the containment cooling trains includes three spray pumps, two spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the SIRWT upon an ESF actuation signal and automatically transferring suction to the containment sump.

The Containment Air Cooler System portion of the containment cooling train which must be OPERABLE includes the three safety related air coolers which each consist of four cooling coil banks, the safety related fan which must be in operation to be OPERABLE, gravity-operated fan discharge dampers, instruments, and controls to ensure an OPERABLE flow path.

CAC fans V-1A, V-2A, V-3A, and V-4A must be in operation to be considered OPERABLE. These fans only receive a start signal from the DBA sequencer; they are assumed to be in operation, and are not started by either a CHP or an SIS signal.

APPLICABILITY

In MODES 1, 2, and 3, a large break LOCA event could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains.

In MODES 4, 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray and Containment Cooling systems are not required to be OPERABLE in MODES 4, 5 and 6.

ACTIONS

A.1

Condition A is applicable whenever one or more containment cooling trains is inoperable. Action A.1 requires restoration of both trains to OPERABLE status within 72 hours. The 72 hour Completion Time for Condition A is based on the assumption that at least 100% of the required post accident containment cooling capability (that assumed in the safety analyses) is available. If less than 100% of the required post containment accident cooling is available, Condition C must also be entered.

Mechanical system LCOs typically provide a 72 hour Completion Time under conditions when a required system can perform its required safety function, but may not be able to do so assuming an additional failure. When operating in accordance with the Required Actions of an LCO

BASES

ACTIONS

Condition, it is not necessary to be able to cope with an additional single failure.

The Containment Cooling systems can provide one hundred percent of the required post accident cooling capability following the occurance of any single active failure. Therefore, the containment cooling function can be met during conditions when those components which could be deactivated by a single active failure are known to be inoperable. Under that condition, however, the ability to provide the function after the occurance of an additional failure cannot be guarranteed. Therefore, continued operation with one or more trains inoperable is allowed only for a limited time.

B.1 and B.2

Condition B is applicable when the Required Actions of Condition A cannot be completed within the required Completion Time. Condition A is applicable whenever one or more trains is inoperable. Therefore, when Condition B is applicable, Condition A is also applicable. (If less than 100% of the post accident containment cooling capability is available, Condition C must be entered as well.) Being in Conditions A and B concurrently maintains both Completion Time clocks for instances where equipment repair allows exit from Condition B while the plant is still within the applicable conditions of the LCO.

If the inoperable containment cooling trains cannot be restored to OPERABLE status within the required Completion Time of Condition A, 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 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

Condition C is applicable with one or more trains inoperable when there is less than 100% of the required post accident containment cooling capability available. Condition A is applicable whenever one or more trains is inoperable. Therefore, when this Condition is applicable, Condition A is also applicable. Being in Conditions A and C concurrently maintains both Completion Time clocks for instances where equipment repair restores 100% of the required post accident containment cooling capability while the LCO is still applicable, allowing exit from Condition C (and LCO 3.0.3).

BASES

ACTIONS

Several specific cases have been analyzed in the safety analysis to provide operating flexibility for equipment outages and testing. These analyses show that action A.1 can be entered under certain circumstances, because 100% of the post accident cooling capability is maintained. These specific cases are discussed below.

One hundred percent of the required post accident cooling capability can be provided with both MSIV bypass valves closed if either;

1. Two containment spray pumps, two spray headers, and one CAC fan are OPERABLE, or

2. One containment spray pump, two spray headers, and three safety related CACs, are OPERABLE (at least two service water pumps must be OPERABLE if CACs are to be relied upon).

One hundred percent of the required post accident cooling capability can be provided for operation with a MSIV bypass valve open or closed if;

1. Two containment spray pumps, two spray headers, and two safety related CACs, are OPERABLE (at least two service water pumps must be OPERABLE if CACs are to be relied upon).

2. One containment spray pump, one spray header, and three safety related CACs are OPERABLE (at least three service water pumps must be OPERABLE to provide the necessary service water flow to assure OPERABILITY of the CACs).

If the Containment Spray side (tube side) of one SDC Heat Exchanger is out of service, 100% of the required post accident cooling capability can be provided, if other equipment outages are limited. One hundred percent of the post accident cooling can be provided with the Containment Spray side of one SDC Heat Exchanger out of service if the following equipment is OPERABLE: 3 safety related Containment Air Coolers, 2 Containment Spray Pumps, 3 CCW pumps, 2 SWS pumps, and both CCW Heat Exchangers, and if

1. One CCW Containment Isolation Valve, CV-0910, CV-0911, or CV-0940, is OPERABLE, <u>and</u>

2. One CCW header isolation valve for the non-safety related loads outside the containment, CV-0944, CV-0944A, or CV-0977B, is OPERABLE.

BASES

ACTIONS

With less than 100% of the required post accident containment cooling capability available, the plant is in a condition outside the assumptions of the safety analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR also does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct positions prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

SR 3.6.6.2

Operating each safety related Containment Air Cooler fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and are functioning properly. The 31 day Frequency was developed considering the known reliability of the fan units, the two train redundancy available, and the low probability of a significant degradation of the containment cooling train occurring between surveillances.

SR 3.6.6.3

Verifying the containment spray header is full of water to the 735 ft elevation minimizes the time required to fill the header. This ensures that spray flow will be admitted to the containment atmosphere within the time frame assumed in the containment analysis. The 31 day Frequency is based on the static nature of the fill header and the low probability of a significant degradation of the water level in the piping occurring between surveillances.

SR 3.6.6.4

Verifying a total service water flow rate of ≥ 4800 gpm to CACs VHX-1, VHX-2, and VHX-3, when aligned for accident conditions, provides assurance the design flow rate assumed in the safety analyses will be achieved (Ref. 8). Also considered in selecting this Frequency were the

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SURVEILLANCE REQUIREMENTS

known reliability of the cooling water system, the two train redundancy, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.5

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5).

Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.6 and SR 3.6.6.7

SR 3.6.6.6 verifies each automatic containment spray valve actuates to its correct position upon receipt of an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. SR 3.6.6.7 verifies each containment spray pump starts automatically on an actual or simulated actuation signal. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power.

Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Where the surveillance of containment sump isolation valves is also required by SR 3.5.2.5, a single surveillance may be used to satisfy both requirements.

SR 3.6.6.8

This SR verifies each containment cooling fan actuates upon receipt of an actual or simulated actuation signal. The 18 month Frequency is

BASES

SURVEILLANCE REQUIREMENTS

based on engineering judgement and has been shown to be acceptable through operating experience. See SR 3.6.6.6 and SR 3.6.6.7, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.6.9

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. Performance of this SR demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.18
3. FSAR, Sections 6.2
4. FSAR, Section 6.3
5. ASME, Boiler and Pressure Vessel Code, Section XI
6. FSAR, Table 14.18.1-3
7. FSAR, Table 14.18.2-1
8. FSAR, Table 9-1
9. EA-MSLB-2001-01 Rev. 0, Containment Response to a MSLB Using CONTEMPT-LT/28, April 2001.
10. EA-LOCA-2001-01 Rev. 0, Containment Response to a LOCA Using CONTEMPT-LT/28, April 2001.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen Recombiners is to eliminate the potential breach of containment due to a sudden hydrogen oxygen reaction. Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and the Palisades Nuclear Plant design criteria (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The recombiners accomplish this by recombining hydrogen and oxygen in a controlled manner to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammability limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiners are provided. Each consists of controls and a power supply located in the auxiliary building, and a recombiner located in containment. The recombiners have no moving parts. When a hydrogen recombiner is placed in operation, containment atmosphere is drawn through the unit by natural convection and the temperature of the air is raised to a level sufficient for recombination of the hydrogen and oxygen to occur (approximately 1150°F). A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.1 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus and is provided with a separate power panel and control panel.

BASES

BACKGROUND (continued)

In order for a Hydrogen Recombiner to be considered OPERABLE, at least one containment cooling safety related fan must be in operation or available for operation. Fan operation is necessary to ensure that the post-accident containment atmosphere is adequately mixed preventing local hydrogen buildups in excess of the flammability limit. The fan must be started or verified to be in operation when the recombiners are placed in operation because these fans do not receive an SIS start signal (although they do receive a DBA sequencer start signal). If there is no qualified fan available, the recombiners would not have all of thier required support equipment and would have to be declared inoperable. Only fans V-1A, V-2A, V-3A, and V-4A (associated with VHX-1, VHX-2, VHX-3, and VHX-4 respectively) are qualified for the post-accident containment environment (i.e., the "B" fans may not be relied upon to provide a post accident function).

LCO 3.6.6, "Containment Cooling Systems," also contains requirements for containment cooling fan OPERABILITY. The restoration time specified for Containment Air Coolers in LCO 3.6.6 is more restrictive than that specified for Hydrogen Recombiners in LCO 3.6.7.

BASES

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analysis are not exceeded and minimizing damage to safety related equipment located in containment. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate within containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the primary coolant;

b. Radiolytic decomposition of water in the Primary Coolant System (PCS) and the containment sump;

c. Hydrogen in the PCS at the time of the LOCA (i.e., hydrogen dissolved in the primary coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to Containment Spray System and Emergency Core Cooling Systems solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions discussed in Reference 3 are used to maximize the amount of hydrogen calculated.

The hydrogen recombiners satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

Two hydrogen recombiners must be OPERABLE. In addition, one safety related containment cooling fan must be in operation or available for operation. These requirements ensure OPERABILITY of at least one hydrogen recombiners and adequate mixing of the containment atmosphere in the event of a worst case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

BASES

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the post LOCA hydrogen concentration within containment below its flammability limit of 4.1 v/o, assuming a worst case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or MSLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or MSLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note stating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one hydrogen recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or MSLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or MSLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

BASES

ACTIONS (continued)

B.1

If the inoperable hydrogen recombiner cannot be restored to OPERABLE status 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. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Performance of a system functional test for each hydrogen recombiner ensures that the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR requires verification that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum for approximately 2 minutes and verified to be ≥ 60 kW. The 18 month Frequency is based on past operating history of the recombiners and engineering judgement.

SR 3.6.7.2

This SR ensures that there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The 18 month Frequency for this SR was developed considering that the incidence of hydrogen recombiners failing the SR in the past is low.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.3

This SR requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms. This SR also requires the performance of a continuity test to ensure there is no single phase fault or any openings of a single heater bank. The continuity test measures the resistance of each phase to neutral. The 18 month Frequency for this SR was developed considering that the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. 10 CFR 50.44
2. FSAR, Section 5.1
3. Regulatory Guide 1.7, Revision 1

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the Primary Coolant Pressure Boundary (PCPB) by providing a heat sink for the removal of energy from the Primary Coolant System (PCS) if the preferred heat sink, provided by the condenser and Circulating Water System, is not available.

Twelve MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 4.3.4 (Ref. 1). The MSSV rated capacity passes the full steam flow at 102% RTP (100% + 2% for instrument error) with twenty-three valves full open. This meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III (Ref. 2). The MSSV design includes staggered lift settings, according to Table 3.7.1-1, in the accompanying LCO, so that only the number of valves needed will actuate. Staggered lift settings reduce the potential for valve chattering because of insufficient steam pressure to fully open all valves following a turbine reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 1; the purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any Anticipated Operational Occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis. (As defined in 10 CFR 50, Appendix A, "Anticipated operational occurances mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.")

BASES

APPLICABLE SAFETY ANALYSES (continued)	The events that challenge the MSSV relieving capacity, and thus PCS pressure, are those characterized as decreased heat removal events, and are presented in the FSAR, Sections 14.12 and 14.13 (Refs. 3 and 4) respectfully. Of these, the full power loss of external load event is the most limiting. The event is initiated by either a loss of external electrical load or a turbine trip. No credit is taken for direct reactor trip on turbine trip, the turbine bypass valve, atmospheric dump valves, or pressurizer PORVs. The reduced heat transfer causes an increase in PCS temperature, and the resulting PCS fluid expansion causes an increase in pressure. The PCS pressure increases to ≤ 2614.9 psia, this peak pressure is < 110% of the design pressure, or 2750 psia for the primary system, with the pressurizer safety valves providing relief capacity. The secondary system pressure increases to 1040.8 psia, this pressure is < 110% of the design pressure, or 1100 psia for the secondary system, with the MSSVs providing relief capability. The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	This LCO requires twenty-three MSSVs to be OPERABLE in compliance with Reference 2. The OPERABILITY of the MSSVs is defined as the ability to open within the lift setting tolerances and to relieve steam generator overpressure. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program. The lift settings, according to Table 3.7.1-1 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure. This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the PCPB.
APPLICABILITY	In MODES 1, 2, and 3 a minimum of twenty-three MSSVs are required to be OPERABLE, to provide overpressure protection required by both ASME Code and the accident analysis. In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

BASES

ACTIONS

A.1

With one or more required MSSVs inoperable, the ability to limit system pressure during accident conditions will be degraded. The four hour Completion Time allows the operator a reasonable amount of time to make minor repairs or adjustments to restore the required number of inoperable MSSVs to OPERABLE status.

B.1 and B.2

If the required MSSVs cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift settings in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 5), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 6). According to Reference 6, the following tests are required for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements.

Table 3.7.1-1 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

The ambient temperature of the operating environment shall be simulated during the set-pressure test in accordance with Reference 6.

REFERENCES

1. FSAR, Section 4.3.4
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components
3. FSAR, Section 14.12
4. FSAR, Section 14.13
5. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWV-1000
6. ANSI/ASME OM-1-1987