

B 3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM Instrumentation is to display unit variables that provide information required by the main control room operators during accident situations. These plant variables provide the necessary information to assess the process of accomplishing or maintaining critical safety functions. The instruments which monitor these variables are designated in accordance with Reference 1.

The OPERABILITY of the PAM Instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. This capability is consistent with the recommendations of Reference 1.

A PAM CHANNEL shall extend from the sensor up to the display device, and shall include the sensor (or sensors), the signal conditioning, any associated datalinks, the display device, any signal gathering or processing subsystems, and any data processing subsystems. Note that for digital PAM CHANNELS, the information may be displayed on multiple display devices. For this case, the PAM CHANNEL shall extend to any available qualified display device.

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category 1 variables. The unit specific implementation of Regulatory Guide 1.97 has not identified any Type A variables, therefore, only Category 1 variables are specified.

APPLICABLE
SAFETY ANALYSES

The PAM Instrumentation ensures that the main control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

PAM Instrumentation that is required in accordance with Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for those monitors which provide information required by the control room operators to assess the process of accomplishing or maintaining critical safety functions. This LCO addresses those Regulatory Guide 1.97 instruments which are listed in Table 3.3.3-1.

The OPERABILITY of the PAM Instrumentation ensures there is sufficient information available on selected plant parameters to monitor and assess plant status following an accident. This capability is consistent with the recommendations of Reference 1.

Category 1 non-type A variables are required to meet Regulatory Guide 1.97 Category 1 (Ref. 1) design and qualification requirements for seismic and environmental qualification, single-failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

Listed below are discussions of the specified instrument functions listed in Table 3.3.3-1. Each of these is a Category 1 variable.

1. Intermediate Range Neutron Flux

Neutron Flux indication is provided to verify reactor shutdown. The neutron flux intermediate range is sufficient to cover the full range of flux that may occur post accident.

Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

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BASES

LCO
(continued)

2, 3. Reactor Coolant System (RCS) Wide Range Hot and Cold Leg Temperature

RCS Hot and Cold Leg Temperatures are provided for verification of core cooling and long-term surveillance. The channels provide indication over a range of [50°F to 700°F].

In addition to this, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the plant conditions necessary to establish natural circulation in the RCS.

4. RCS Pressure

RCS wide range pressure is provided for verification of core cooling and RCS integrity long term surveillance.

5. Pressurizer Pressure and RCS Subcooling Monitor

Pressurizer Pressure is used to determine RCS Subcooling. The RCS Subcooling Monitor is provided for verification of core cooling. Subcooling margin is available when the RCS pressure is greater than the saturation pressure corresponding to the core exit temperature. Inputs to the Subcooling Monitor are pressurizer pressure and RCS hot leg temperature.

6. Containment Water Level

Containment Water Level is used to monitor the containment environment during accident conditions. The containment water level can also provide information to the operators that the various stages of safety injection along with system depressurization are progressing.

7. Containment Pressure

The containment pressure transmitters monitor the containment pressure over the range of [-5 to 10 psig]. This provides information on post accident containment pressure and containment integrity.

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BASES

LCO
(continued)

8. Containment Pressure (Extended Range)

The extended range containment pressure transmitters are instruments that operators use for monitoring the potential for breach of containment, a fission product barrier. The extended range sensors monitor containment pressure over the range of [0 to 240 psig].

9. Containment Area Radiation (High Range)

Containment Area Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

10. Pressurizer Level and Associated Reference Leg Temperature

Pressurizer level is provided to monitor the RCS coolant inventory. During an accident, operation of the safeguards systems can be verified based on coolant inventory indicators.

The reference leg temperature is included in the Technical Specification since it is used to compensate the level signal.

11. In-Containment Refueling Water Storage Tank (IRWST) Water Level

The IRWST provides a long term heat sink for non-LOCA events and is a source of injection flow for LOCA events. When the IRWST is a heat sink, the level will change due to increased volume associated with the temperature increase. When saturation temperature is reached, the IRWST will begin steaming and initially lose mass to the containment atmosphere until condensation occurs on the steel containment shell which is cooled by the passive containment cooling system. The condensate is returned to the IRWST via a gutter.

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BASES

LCO

11. In-Containment Refueling Water Storage Tank (IRWST) Water Level (continued)

During a LOCA, the IRWST is available for injection. Depending on the severity of the event, when a fully depressurized RCS has been achieved, the IRWST will inject by gravity flow.

12. Passive Residual Heat Removal (PRHR) Flow and PRHR Outlet Temperature

PRHR Flow is provided to monitor primary system heat removal during accident conditions when the steam generators are not available. PRHR provides primary protection for non-LOCA events when the normal heat sink is lost.

PRHR outlet temperature is provided to monitor primary system heat removal during accident conditions when the steam generators are not available. PRHR provides primary protection for non-LOCA events when the normal heat sink is lost.

13, 14, 15, 16. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

An evaluation was made of the minimum number of valid core exit thermocouples necessary for In-Core Cooling (ICC) detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate ICC detection is assured with two valid core exit thermocouples per quadrant. Core Exit Temperature is also used for plant stabilization and cooldown monitoring.

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BASES

LCO 13, 14, 15, 16. Core Exit Temperature (continued)

Two OPERABLE channels of Core Exit Temperature are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Two thermocouples in each of the two divisions ensure a single failure will not disable the ability to determine the temperature at two locations within a quadrant.

17. Passive Containment Cooling System (PCS) Storage Tank Level and PCS Flow

The PCS must be capable of removing the heat from the containment following a postulated LOCA or steam line break (SLB). The tank level instruments provide indication that sufficient water is available to meet this requirement. The PCS flow instrument provides a diverse indication of the PCS heat removal capability.

18. Remotely Operated Containment Isolation Valve Position

The Remotely Operated Containment Isolation Valve Position is provided for verification of containment OPERABILITY.

19. IRWST to RNS Suction Valve Status

The position of the motor-operated valve in the line from the IRWST to the pump suction header is monitored to verify that the valve is closed following postulated events. The valve must be closed to prevent loss of IRWST inventory into the RNS.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables provide the information necessary to assess the process of accomplishing or maintaining critical safety functions following Design Basis Accidents (DBAs). The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, plant conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

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BASES (continued)

ACTIONS

The ACTIONS Table has been modified by two Notes.

The first Note excludes the MODE change restriction 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 exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident using alternate instruments and methods, and low probability of an event requiring these instruments.

The second Note in the ACTIONS clarifies the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that function.

A.1

When one or more Functions have one required channel which is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

The Required Action directs actions to be taken in accordance with Specification 5.6.7 immediately. Each time an inoperable channel has not met Required Action A.1, and the associated Completion Time has expired, Condition B is entered.

C.1

When one or more Functions have two required channels which are inoperable, (two channels inoperable in the same Function), one channel in the Function should be restored to

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BASES

ACTIONS

C.1 (continued)

OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information.

Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM function will be in a degraded condition should an accident occur.

D.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.3-1. The applicable Condition referenced in the Table is Function dependent.

Each time an inoperable channel has not met any Required Action of Condition C, and the associated Completion Time has expired, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C are not met for the Functions in Table 3.3.3-1, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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BASES (continued)

SURVEILLANCE
REQUIREMENTS

The following SRs apply to each PAM instrumentation function in Table 3.3.3-1:

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days verifies that a gross instrumentation failure has not occurred. A CHANNEL CHECK is 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 that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar plant instruments located throughout the plant.

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

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of those displays associated with the required channels of this LCO.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop including the sensor. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. This SR is modified by a Note that excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." RTD and Thermocouple channels are to be calibrated in place using cross-calibration techniques. The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. Regulatory Guide 1.97, Rev. 3, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," U.S. Nuclear Regulatory Commission.
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B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown Workstation (RSW)

BASES

BACKGROUND

The RSW provides the control room operator with sufficient displays and controls to place and maintain the unit 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. Passive residual heat removal (PRHR), the core makeup tanks (CMTs), and the in-containment refueling water storage tank (IRWST) can be used to remove core decay heat. The use of passive safety systems allows extended operation in MODE 4.

If the control room becomes inaccessible, the operators can establish control at the RSW and place and maintain the unit in MODE 4 with $T_{avg} < 350^{\circ}\text{F}$. The unit can be maintained safely in MODE 4 with $T_{avg} < 350^{\circ}\text{F}$ for an extended period of time.

The OPERABILITY of the remote shutdown control and display functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 4 with $T_{avg} < 350^{\circ}\text{F}$ should the control room become inaccessible.

APPLICABLE
SAFETY ANALYSES

The RSW is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 4 with $T_{avg} < 350^{\circ}\text{F}$.

The criteria governing the design and the specific system requirements of the RSW are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

Since the passive safety systems alone can establish and maintain safe shutdown conditions for the unit, nonsafety systems are not required for safe shutdown of the unit. Therefore, no credit is taken in the safety analysis for nonsafety systems.

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The RSW satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The RSW LCO provides the OPERABILITY requirements of the displays and controls necessary to place and maintain the unit in MODE 4 from a location other than the control room.

The RSW is OPERABLE if the display instrument and control functions needed to support the RSW are OPERABLE.

The RSW covered by this LCO does not need to be energized to be considered OPERABLE. This LCO is intended to ensure the RSW will be OPERABLE if unit conditions require that the RSW be placed in operation.

APPLICABILITY

The RSW LCO is applicable in MODES 1, 2, and 3 and in MODE 4 with $T_{avg} < 350^{\circ}\text{F}$. This is required so that the facility can be placed and maintained in MODE 4 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4 with $T_{avg} < 350^{\circ}\text{F}$ or in MODE 5 or 6. In these MODES, the unit is already subcritical and in a condition of reduced Reactor Coolant System (RCS) energy. Under these conditions, considerable time is available to restore necessary instrument control functions if control room instruments or controls become unavailable.

ACTIONS

The Note excludes the MODE change restriction 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 unit shutdown. This exception is acceptable due to the low probability of an event requiring the RSW and because the equipment can generally be repaired during operation without significant risk of a spurious trip.

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BASES

ACTIONS
(continued)

A.1

Condition A addresses the situation where the RSW is inoperable. The Required Action is to restore the RSW to OPERABLE status within 30 days. 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 is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 with $T_{avg} < 350^{\circ}\text{F}$ within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 verifies that each required RSW transfer switch performs the required functions. This ensures that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 4 with $T_{avg} < 350^{\circ}\text{F}$ from the RSW. The 24 month Frequency was developed considering it is prudent that these types of surveillances be performed during a unit outage. However, this surveillance is not required to be performed only during a unit outage. This is due to the unit conditions needed to perform the surveillance and the potential for unplanned transients if the surveillance is performed with the reactor at power. Operating experience demonstrates that RSW transfer switches usually pass the surveillance test when performed on the 24 month Frequency.

SR 3.3.4.2

This Surveillance verifies that the RSW communicates controls and indications with Divisions A, B, C, and D of the PMS. Communication is accomplished by use of separate multiplexers for each division. The operator can select the controls and indications available through each PMS division.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.4.2 (continued)

The Frequency is based on the known reliability of the Functions and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.4.3

SR 3.3.4.3 verifies the OPERABILITY of the RSW hardware and software by performing diagnostics to show that operator displays are capable of being called up and displayed to an operator at the RSW. The RSW has several video display units which can be used by the operator. The video display units are identical to that provided in the control room and the operator can display information on the video display units in a manner which is identical to the way the information is displayed in the control room. The operator normally selects an appropriate set of displays based on the particular operational goals being controlled by the operator at the time. Each display consists of static graphical and legend information which is contained within the display processor associated with each video display unit and dynamic data which is updated by the data display system.

The Frequency of 24 months is based on the use of the data display capability in the control room as part of the normal unit operation and the availability of multiple video display units at the RSW. The Frequency of 24 months is based upon operating experience and consistency with control room hardware and software.

SR 3.3.4.4

SR 3.3.4.4 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) every 24 months. This test should verify the OPERABILITY of the reactor trip breakers (RTBs) open and closed indication on the RSW by actuating the RTBs. The Frequency of 24 months was chosen because the RTBs may not be exercised while the facility is at power and is based on operating experience and consistency with the refueling outage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.
 2. Section 7.4.1, "Safe Shutdown."
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B 3.3 INSTRUMENTATION

B 3.3.5 Diverse Actuation System (DAS) Manual Controls

BASES

BACKGROUND

The Diverse Actuation System (DAS) manual controls provide backup controls in case of common mode failure of the Protection and Safety Monitoring System (PMS) automatic and manual actuations evaluated in the AP1000 PRA. These controls are not credited for mitigating accidents in the DCD Chapter 15 analyses.

The specific DAS controls were selected based on PRA risk importance as discussed in Reference 1. As noted in Reference 1, electrical power for these controls and instrument indications need not be covered by Technical Specifications. The rationale is that these controls use the same nonsafety-related power supply used by the plant control system. This power is required to be available to support normal operation of the plant. With offsite power available, there are several sources to provide this power including AC power to non-Class 1E battery chargers, AC power to rectifiers, and non-Class 1E batteries. As a result, with offsite power available it is very likely that power will be available for these DAS controls. If offsite power is not available, then there is still the likelihood that the non-1E batteries or the non-1E diesel generators will be available. Even if these sources are unavailable, the desired actions will occur without operator action for the more probable events. The rods will insert automatically on loss of offsite power. The passive residual heat removal heat exchanger (PRHR HX), core makeup tanks (CMT), passive containment cooling system (PCS), and containment isolation features are initiated by operation of fail-safe, air-operated valves. If all offsite and onsite AC power is lost, the instrument air system will depressurize by the time these functions are needed in the 1-hour time frame.

Instrument readouts are expected to be available even in case of complete failure of the PMS due to common cause failure. These instruments include both DAS and PLS instruments. As discussed above, it is expected that AC power will be available to power the instruments. Even if the operators have no instrument indications, they are

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BASES

BACKGROUND
(continued)

expected to actuate the controls most likely to be needed (PRHR HX, CMT, PCS, and containment isolation). If all AC power fails, then the rods will drop and the air-operated valves will go to their fail-safe positions.

The DAS uses equipment from sensor output to the final actuated device that is diverse from the PMS to automatically initiate a reactor trip, or to manually actuate the identified safety-related equipment. DCD Section 7.7.1.11 (Ref. 2) provides a description of the DAS.

APPLICABLE
SAFETY ANALYSES

The DAS manual controls are required to provide a diverse capability to manually trip the reactor and actuate the specified safety-related equipment, based on risk importance in the AP1000 PRA.

The DAS manual controls are not credited for mitigating accidents in the DCD Chapter 15 safety analyses.

The AP1000 PRA, Appendix A, provides additional information, including the thermal and hydraulic analyses of success sequences used in the PRA.

The DAS manual controls satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The DAS LCO provides the requirements for the OPERABILITY of the DAS manual trip and actuation controls necessary to place the reactor in a shutdown condition and to remove decay heat in the event that the PMS automatic actuation and manual controls are inoperable.

APPLICABILITY

The DAS manual controls are required to be OPERABLE in the MODES specified in Table 3.3.5-1.

The manual DAS reactor trip control is required to be operable in MODES 1 and 2 to mitigate the effects of an ATWS event occurring during power operation.

The other manual DAS actuation controls are required to be available in the plant MODES specified, based on the need

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BASES

APPLICABILITY (continued) for operator action to actuate the specified components during events that may occur in these various plant conditions, as identified in the AP1000 PRA.

ACTIONS

A.1

Condition A applies when one or more DAS manual controls are inoperable.

The Required Action A.1 to restore the inoperable DAS manual control(s) to OPERABLE status within 30 days is reasonable because the DAS is a separate and diverse non-safety backup system for the manual reactor trip and manual safety-related equipment actuation controls. The 30-day Completion Time allows sufficient time to repair an inoperable manual DAS control but ensures the control is repaired to provide backup protection.

B.1 and B.2

Condition B applies when Required Action A cannot be completed for the DAS manual reactor trip control within the required completion time of 30 days.

Required Action B.1 requires SR 3.3.1.5, "Perform TADOT" for the reactor trip breakers, is to be performed once per 31 days, instead on once every 92 days. The predominant failure requiring the DAS manual reactor trip control is common mode failure of the reactor trip breakers. This change in surveillance frequency for testing the reactor trip breakers increases the likelihood that a common mode failure of the reactor trip breakers would be detected while the DAS manual reactor trip control is inoperable. This reduces the likelihood that a diverse manual reactor trip is required. It is not required to perform a TADOT for the manual actuation control. The manual reactor trip control is very simple, highly reliable, and does not use software in the circuitry.

Action B.2 requires that the inoperable DAS manual reactor trip control be restored to OPERABLE status prior to entering MODE 2 following any plant shutdown to MODE 5 while the control is inoperable. This ACTION is provided to ensure that all DAS manual controls are restored to OPERABLE status following the next plant shutdown.

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BASES

ACTIONS
(continued)C.1 and C.2

Condition C applies when Required Action A cannot be completed for any DAS manual actuation control (other than reactor trip) within the required completion time of 30 days.

Required Action C.1 requires SR 3.3.2.2, "Perform ACTUATION LOGIC TEST," to be performed once per 31 days, instead on once every 92 days. The predominant failure requiring the DAS manual reactor trip control is common mode failure of the PMS actuation logic software or hardware. This change in surveillance frequency for actuation logic testing increases the likelihood that a common mode failure of the PMS actuation logic from either cause would be detected while any DAS manual actuation control is inoperable. This reduces the likelihood that a diverse component actuation is required. It is not required to perform a TADOT for the manual actuation control device since the manual actuation control devices are very simple and highly reliable.

Action C.2 requires that the inoperable DAS manual actuation control(s) be restored to OPERABLE status prior to entering MODE 2 following any plant shutdown to MODE 5 while the control is inoperable. This ACTION is provided to ensure that all DAS manual controls are restored to OPERABLE status following the next plant shutdown.

D.1 and D.2

Condition D is entered if the Required Action associated with Condition B or C is not met within the required Completion Time.

Required Actions D.1 and D.2 ensure that the plant is placed in a condition where the probability and consequences of an event are minimized. The allowed Completion Times are reasonable based on plant operating experience, for reaching the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

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BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.1

SR 3.3.5.1 is the performance of a TADOT of the DAS manual trip and actuation controls for the specified safety-related equipment. This TADOT is performed every 24 months.

The Frequency is based on the known reliability of the DAS functions and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of the setpoints from the TADOT. The functions have no setpoints associated with them.

REFERENCES

1. WCAP-15985, "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," November 2002.
 2. DCD, Section 7.7.1.11.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within the 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 of operating conditions. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

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

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

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. At the beginning of each fuel cycle, precision (calorimetric) flow measurements provide a value for comparison to the limit. The cold leg flow rate channels are normalized to the calorimetric flow measurement for 100% indication and are frequently monitored to determine flow degradation. A lower RCS flow will cause the core to approach DNB limits.

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

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BASES (continued)

APPLICABLE
SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown transients initiated within the requirements of this LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit which could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed include loss of coolant flow events and dropped or stuck rod events. An assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit and the RCS average temperature limit specified in the COLR correspond to analytical limits, with an allowance for steady state fluctuations and measurement errors. The RCS average temperature limit corresponds to the analytical limit with allowance for controller deadband and measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables, pressurizer pressure, RCS average temperature, and RCS total flow rate to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, usually based on [maximum analyzed steam generator tube plugging], is retained in the TS LCO. Operating within these limits will result in meeting DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error based on performing a precision heat balance and using the result to normalize the RCS flow rate indicators.

The numerical values for pressure, temperature, and flow rate specified in the COLR are given for the measurement location but have been adjusted for instrument error.

(continued)

BASES (continued)

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

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether an SL may have been exceeded.

ACTIONS

A.1

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

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

(continued)

BASES

ACTIONS

A.1 (continued)

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

B.1

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

SURVEILLANCE
REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency of pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

(continued)

BASES

**SURVEILLANCE
REQUIREMENTS**
(continued)

SR 3.4.1.3

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

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 24 months, at the beginning of each fuel cycle, allows the installed RCS flow instrumentation to be normalized and verifies the actual RCS flow is greater than or equal to the minimum required RCS flow rate.

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

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until after 24 hours after $\geq 90\%$ RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP.

REFERENCES

1. Chapter 15, "Accident Analyses."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

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

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

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

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

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

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

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

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

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

LCO

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

APPLICABILITY

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

The special test exception of LCO 3.1.8, "MODE 2 PHYSICS TEST Exceptions," permits PHYSICS TESTS to be performed at $\leq 5.0\%$ RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for

(continued)

BASES

APPLICABILITY
(continued)

nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

ACTIONS

A.1

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

SURVEILLANCE
REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above [551]°F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

REFERENCES

1. Chapter 15, "Accident Analyses."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

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

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

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

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

10 CFR 50, Appendix G (Ref. 1) requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. An adequate margin to brittle failure must be provided during normal operation, anticipated operational occurrences, and system hydrostatic tests. Reference 1 mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

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

(continued)

BASES

BACKGROUND
(continued)

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. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 5).

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 P/T span of the limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

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

The criticality limit curve includes the Reference 1 requirement that it be $\geq 40^\circ\text{F}$ above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH Testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 6) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 7 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

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

LCO

The two elements of this LCO are:

- a. The limit curves for heatup, cooldown, ISLH testing and criticality; and
- b. Limits on the rate of change of temperature.

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

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

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

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

(continued)

BASES

LCO
(continued)

- 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.
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APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile (brittle) failure in accordance with 10 CFR 50, Appendix G (Ref. 1). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, they are applicable at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

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.

A.1 and A.2

Operation outside the P/T limits must be restored to within the limits. The RCPB must be returned to a condition that has been verified by stress analyses. Restoration is in the proper direction to reduce RCPB stress.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with preanalyzed 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.

Condition A is modified by a Note requiring Required Action A.2 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 per Required Action A.1 alone is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

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

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 4 within 24 hours, with RCS pressure < 500 psig.

The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power condition in an orderly manner without challenging plant systems.

C.1 and C.2

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

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

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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.

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

SURVEILLANCE
REQUIREMENTS

SR 3.4.3.1

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

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

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

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BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix G, "Fracture Toughness Requirements."
 2. ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure."
 3. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.
 4. 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."
 5. "Embrittlement of Reactor Vessel Materials," May 1988.
 6. ASME Boiler and Pressure Vessel Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events."
 7. WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops

BASES

BACKGROUND

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

The secondary functions of the RCS include:

- a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;
- b. Improving the neutron economy by acting as a reflector;
- c. Carrying the soluble neutron poison, boric acid;
- d. Providing a second barrier against fission-product release to the environment; and
- e. Removal of the heat generated in the fuel due to fission-product decay following a unit shutdown.

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

The RCPs are powered by variable speed controllers when the RCS coolant temperature is below 450°F. The pumps must be started using the variable speed controller with the reactor trip breakers open. The controller shall be bypassed prior to closure of the reactor trip breakers.

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BASES (continued)APPLICABLE
SAFETY ANALYSESMODES 1 and 2

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

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming two RCS loops are initially in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses, where RCP operation is most important are the four pump coastdown, single pump locked rotor, single pump broken shaft or coastdown, and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the two RCS loop operation. For two RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 102% RATED THERMAL POWER (RTP). This is the design overpower condition for two RCS loop operation. The value for the accident trip setpoint of the nuclear overpower (high flux) trip is 118% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points which result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

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

(continued)

BASESAPPLICABLE
SAFETY ANALYSES
(continued)MODES 3, 4, and 5

Whenever the reactor trip breakers are in the closed position and the control rod drive mechanisms (CRDMs) are energized, there is the possibility of an inadvertent rod withdrawal from subcritical, resulting in a power excursion in the area of the withdrawn rod. Such a transient could be caused by a malfunction of the Plant Control System (PLS). In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM. The initial power rise is terminated by doppler broadening in the fuel pins, followed by rod insertion. During this event, if there is not adequate coolant flow along the clad surface of the fuel, there is a potential to exceed the departure from nucleate boiling ratio (DNBR) limit. Therefore, the required coolant flow is an initial condition of a design basis event that presents a challenge to the integrity of a fission product barrier.

Therefore, in MODE 3, 4 or 5 with the RTBs in the closed position and the PLS capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires the RCPs to be OPERABLE and in operation to ensure that the accident analysis limits are met.

In MODES 3, 4 and 5 with the RTBs open, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. This is addressed in LCO 3.4.9, "Minimum RCS Flow."

RCS Loops satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required in MODES 1 and 2. The requirement that at least four RCPs must be operating in MODES 3, 4 and 5 when the RTBs are closed provides assurance that, in the event of a rod withdrawal accident, there will be adequate

(continued)

BASES

LCO
(continued)

flow in the core to avoid exceeding the DNBR limit. Bypass of the RCP variable speed control ensures that the pumps are operating at full flow.

With the RTBs in the open position, the PLS is not capable of rod withdrawal; therefore only a minimum RCS flow of 10,000 gpm is necessary to ensure removal of decay heat from the core in accordance with LCO 3.4.9, Minimum RCS Flow.

Note 1 prohibits startup of a RCP when the reactor trip breakers are closed. This requirement prevents startup of a RCP and the resulting circulation of cold and/or unborated water from an inactive loop into the core, precluding reactivity excursion events which are unanalyzed.

Note 2 requires that the secondary side water temperature of each SG be $\leq [50]^{\circ}\text{F}$ above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature $\leq 275^{\circ}\text{F}$. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 3 permits all RCPS to be de-energized in MODE 3, 4, or 5 for ≤ 1 hour per 8 hour period. The purpose of the NOTE is to permit tests that are designed to validate various accident analysis values. One of these tests is for the validation of the pump coastdown curve, used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve may need to be revalidated by conducting the test again.

Another test performed during the startup testing program is the validation of the rod drop times during cold conditions, both with and without flow.

The no-flow tests may be performed in MODE 3, 4, or 5, and require that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should only be performed once, unless the flow characteristics of the RCS are changed. The 1 hour time

period specified is adequate to perform the desired tests and experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the NOTE is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause natural circulation flow obstruction.

An OPERABLE RCS loop is composed of two OPERABLE RCPs in operation providing forced flow for heat transport and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, Section 5.5.5.

APPLICABILITY

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

In MODES 3, 4 and 5, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. For these purposes and because the reactor trip breakers are closed, there is the possibility of an inadvertent rod withdrawal event. Four RCPs are required to be operating in MODES 3, 4 and 5, whenever the reactor trip breakers are closed.

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BASES (continued)

ACTIONS

A.1

If the requirements of the LCO are not met while in MODE 1 or 2, the Required Action is to reduce power and bring the plant to MODE 3 with the reactor trip breakers open. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

Condition A is modified by a Note which requires completion of Required Action A.1 whenever the Condition is entered. This ensures that no attempt is made to restart a pump with the reactor trip breakers closed, thus precluding events which are unanalyzed.

When all four reactor coolant pumps are operating, a loss of a single reactor coolant pump above power level P-8 will result in an automatic reactor trip. When three reactor coolant pumps are operating, a loss of a single reactor coolant pump above power level P-10 will result in an automatic reactor trip.

The Completion Time of 6 hours is reasonable to allow for an orderly transition to MODE 3. The applicable safety analyses described above bound Design Basis Accidents (DBA) initiated with three reactor coolant pumps operating at power levels below P-8, and with two reactor coolant pumps at power levels below P-10.

B.1

If the requirements of the LCO are not met while in MODE 3, 4 or 5, the Required Action is to remain in MODE 3, 4 or 5 and open the reactor trip breakers. This action eliminates the possibility of a rod withdrawal event with one or more pumps not operating and thus minimizing the possibility of violating DNB limits.

Condition B is modified by a Note which requires completion of Required Action B.1 whenever the Condition is entered. This ensures that no attempt is made to restart a pump with the reactor trip breakers closed, thus precluding events which are unanalyzed.

(continued)

BASES

ACTIONS

B.1 (continued)

The Completion Time of 1 hour is reasonable to allow for planned opening of the reactor trip breakers, since plant cool-down is not required.

SURVEILLANCE
REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours that each RCS loop is in operation with the pump variable speed control bypassed. Verification includes flow rate and temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the main control room to monitor RCS loop performance.

REFERENCES

1. Chapter 15, "Accident Analysis."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 Not Used.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 Pressurizer

BASES

BACKGROUND

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

The normal level and pressure control components addressed by this LCO include the pressurizer water level, the heaters, their controls, and power supplies. Pressurizer safety valves and automatic depressurization valves are addressed by LCO 3.4.7, "Pressurizer Safety Valves," and LCO 3.4.12, "Automatic Depressurization System (ADS) - Operating," respectively.

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

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure.

APPLICABLE
SAFETY ANALYSES

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

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Safety analyses presented in Chapter 15 (Ref. 1) do not take credit for pressurizer heater operation, however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirement for the pressurizer water volume $\leq 92\%$ of span, ensures that an adequate steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

APPLICABILITY

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

ACTIONS

A.1 and A.2

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions.

If the pressurizer water level is above the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. This is done by restoring the level to within limit, within 6 hours, or by placing the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

unit in MODE 3 with the reactor trip breakers open within 6 hours, and placing the unit in MODE 4 within 12 hours. This takes the unit out of the applicable MODES and restores the unit to operation within the bounds of the safety analyses.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.6.1

This SR requires that during steady state operation, pressurizer 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. The Frequency of 12 hours corresponds to verifying the parameter each shift. 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.

REFERENCES

1. Chapter 15, "Accident Analysis."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 Pressurizer Safety Valves

BASES

BACKGROUND

The two pressurizer safety valves provide, in conjunction with the Protection and Safety Monitoring System (PMS), overpressure protection for the RCS. The pressurizer safety valves are totally enclosed, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2733.5 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The minimum relief capacity for each valve, 750,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The pressurizer safety valves discharge into the containment atmosphere. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 when the reactor vessel head is on; however, in MODE 4 with the RNS aligned, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.15, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the $\pm 1\%$ tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

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

(continued)

BASES

BACKGROUND
(continued)

The consequences of exceeding the ASME Code, Section III pressure limit (Ref. 1) could include damage to RCS components, increased LEAKAGE, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE
SAFETY ANALYSES

All accident and safety analyses in Chapter 15 (Ref. 3) that require safety valve actuation assume operation of two pressurizer safety valves to limit increases in the RCS pressure. The overpressure protection analysis (Ref. 2) is also based on operation of the two safety valves. Accidents that could result in overpressurization if not properly terminated include:

- a. Uncontrolled rod withdrawal from full power;
- b. Loss of reactor coolant flow;
- c. Loss of external electrical load;
- d. Locked rotor; and
- e. Loss of AC power/loss of normal feedwater

Detailed analyses of the above transients are contained in Reference 3. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer Safety Valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psia), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the $\pm 1\%$ tolerance requirements (Ref. 1) for lifting pressures above 1000 psig.

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

(continued)

BASES (continued)

APPLICABILITY In MODES 1, 2, and 3, and portions of MODE 4 with the RNS isolated or with the RCS temperature $\geq 275^{\circ}\text{F}$, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 with RNS open and in MODE 5, because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

The Note allows entry into MODES 3 and 4 with the lift setpoints outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

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

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two pressurizer safety valves are inoperable, the plant must be placed in a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with the RNS aligned to the RCS and RCS temperature $< 275^{\circ}\text{F}$ within 24 hours.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With the RNS aligned to the RCS, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

SURVEILLANCE
REQUIREMENTS

SR 3.4.7.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested one at a time and in accordance with the requirements of ASME Code Section XI (Ref. 4), which provides the activities and Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is $\pm 3\%$ for OPERABILITY; however, the values are reset to $\pm 1\%$ during the Surveillance to allow for drift.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, NB 7614.3.
 2. [WCAP-7769, "Topical Report on Overpressure Protection, October 1971."]
 3. Chapter 15, "Accident Analyses."
 4. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Operational LEAKAGE

BASES

BACKGROUND

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

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

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

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

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

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

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address 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 a 300 gpd primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) 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 leak contaminates the secondary fluid.

The Chapter 15 (Ref. 3) analyses for the accidents involving secondary side releases assume 150 gpd primary to secondary LEAKAGE in each generator as an initial condition. The design basis radiological consequences resulting from a postulated SLB accident and SGTR are provided in Sections 15.1.5 and 15.6.3 of Chapter 15, respectively.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operation LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

b. Unidentified LEAKAGE

0.5 gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air N13/F18 radioactivity monitoring and containment sump level monitoring equipment, can

(continued)

BASES

LCO

b. Unidentified LEAKAGE (continued)

detect within a reasonable time period. This leak rate supports leak before break (LBB) criteria. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE. Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Both Steam Generators (SGs)

Total primary to secondary LEAKAGE through both SGs amounting to 300 gpd produces acceptable offsite doses in the Steam Line Break (SLB) accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through One SG

The 150 gpd limit from one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture.

The 150 gpd leakage limit is consistent with guidance provided in Reference 4.

f. Primary to IRWST LEAKAGE through the PRHR Heat Exchanger (HX)

The 500 gpd limit from the PRHR HX is based on the assumption that a single crack leaking this amount

(continued)

BASES

LCO

f. Primary to IRWST LEAKAGE through the PRHR Heat Exchanger (HX) (continued)

would not lead to a PRHR HX tube rupture under the stress condition of an RCS pressure increase event. If leaked through many cracks, the cracks are very small, and the above assumption is conservative. This is conservative because the thickness of the PRHR HX tubes is approximately 60% greater than the thickness of the SG tubes. Furthermore, a PRHR HX tube rupture would result in an isolable leak and would not lead to a direct release of radioactivity to the atmosphere.

APPLICABILITY

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

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

ACTIONS

A.1

Unidentified LEAKAGE, 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 RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, 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. It should be noted that leakage past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

36 hours. This action reduces the LEAKAGE and also reduces the factors which tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without ACTIONS challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE
REQUIREMENTS

SR 3.4.8.1

Verifying RCS LEAKAGE within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection.

Unidentified LEAKAGE and identified LEAKAGE are determined by performance of a RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be met with the reactor at steady state operating conditions. Therefore, a Note is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, and with no makeup or letdown.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere N13/F18 radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.8.1 (continued)

boundary LEAKAGE. These LEAKAGE detection systems are specified in LCO 3.4.10, "RCS LEAKAGE Detection Instrumentation."

The containment atmosphere N13/F18 radioactivity leakage measurement is valid only for plant power > 20% RTP.

The containment atmosphere N13/F18 radioactivity leakage measurement during MODE 1 is not valid while containment purge occurs or within 2 hours after the end of containment purge.

The containment sump level change method of detecting leaks during MODES 1, 2, 3, and 4 is not valid while containment purge occurs or within 2 hours after the end of containment purge.

The containment sump level change method of detecting leaks during MODES 1, 2, 3, and 4 is not valid during extremely cold outside ambient conditions when frost is forming in the interior of the containment vessel.

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

SR 3.4.8.2

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. 10 CFR 50, Appendix A GDC 30.
 2. Regulatory Guide 1.45, May 1973.
 3. Chapter 15, "Accident Analysis."
 4. NEI-97-06 - "Steam Generator Program Guidelines."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Minimum RCS Flow

BASES

BACKGROUND

The AP1000 RCS consists of the reactor vessel and two heat transfer loops, each containing a steam generator (SG), two reactor coolant pumps (RCPs), a single hot leg and two cold legs for circulating reactor coolant. Loop 1 also contains connections to the pressurizer and passive residual heat removal (PRHR).

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

Within the RCS, coolant loop flow can be provided by the reactor coolant pumps, the Normal Residual Heat Removal System (RNS), and to a lesser degree when in the passive mode of operation, natural circulation.

APPLICABLE SAFETY ANALYSES

An initial condition in the Design Basis Accident (DBA) analysis of a possible Boron Dilution Event (BDE) in MODE 3, 4, or 5 is the assumption of a minimum mixing flow in the RCS. In this scenario, dilute water is inadvertently introduced into the RCS, is uniformly mixed with the primary coolant, and flows to the core. The increase in reactivity is detected by the source range instrumentation which provides a signal to terminate the inadvertent dilution before the available SHUTDOWN MARGIN (SDM) is lost. If there is inadequate mixing in the RCS, the dilute water may stratify in the primary system, and there will be no indication by the source range instrumentation that a dilution event is in progress. When primary flow is finally increased, the dilution event may have progressed to the point that mitigation by the source range instrumentation is too late to prevent the loss of SDM.

Thus, a minimum mixing flow in the RCS is a process variable which is an initial condition in a DBA analysis.

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Minimum RCS Flow satisfies Criterion 2 of 10 CFR
50.36(c)(2)(ii).

LCO

The requirement that a minimum RCS flow be maintained provides assurance that in the event of an inadvertent BDE, the diluted water will be properly mixed with the primary system coolant, and the increase in core reactivity will be detected by the source range instrumentation.

NOTE 1 permits all RCPS to be de-energized for ≤ 1 hour per 8 hour period. The purpose of the NOTE is to permit tests that are designed to validate various accident analysis values. One of these tests is for the validation of the pump coastdown curve, used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve may need to be revalidated by conducting the test again.

Another test performed during the startup testing program is the validation of the rod drop times during cold conditions, both with and without flow.

The no-flow tests may be performed in MODE 3, 4, or 5, and require that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should only be performed once, unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests and experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the NOTE is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation and

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

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

APPLICABILITY

Minimum RCS flow is required in MODES 3, 4, and 5 with the reactor trip breakers (RTBs) open because an inadvertent BDE is considered possible in these MODES.

In MODES 1 and 2, and in MODES 3, 4, and 5 with the RTBs closed, LCO 3.4.4 requires all four RCPs to be in operation. Thus, in the event of an inadvertent boron dilution, adequate mixing will occur.

A minimum mixing flow is not required in MODE 6 because LCO 3.9.2 requires that all valves used to isolate unborated water sources shall be secured in the closed position. In this situation, an inadvertent BDE is not considered credible.

ACTIONS

A.1

If no RCP is in operation, all sources of unborated water must be isolated within 1 hour. This action assures that no unborated water will be introduced into the RCS when proper mixing cannot be assured. The allowed Completion Time requires that prompt action be taken, and is based on the low probability of a DBA occurring during this time.

A.2

The Requirement to perform SR 3.1.1.1 (SDM verification) within 1 hour assures that if the boron concentration in the RCS has been reduced and not detected by the source range instrumentation, prompt action may be taken to restore the required SDM. The allowed Completion Time is consistent with that required of Action A.1 because the conditions and consequences are the same.

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BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.9.1

This Surveillance requires verification every 12 hours that a minimum mixing flow is present in the RCS. A Frequency of 12 hours is adequate considering the low probability of an inadvertent BDE during this time, and the ease of verifying the required RCS flow.

A minimum mixing flow is provided if any of the following conditions are met:

<u>No. of Pumps Operating</u>	<u>% Rated Speed (each pump)</u>
1	25%
2	20%
3	15%
4	10%

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10CFR50 (Ref. 1) requires means for detecting, and, to the extent practical, identifying the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting LEAKAGE detection systems.

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

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

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity used for leak detection is the decay of N13/F18. The production of N13 and F18 is proportional to the reactor power level. N13 has a short half life and comes to equilibrium quickly. F18 has a longer half life and is the dominant source used for leak detection. Instrument sensitivities for gaseous monitoring are practical for these LEAKAGE detection systems. The Radiation Monitoring System includes monitoring N13/F18 gaseous activities to provide leak detection.

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BASES (continued)

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 response times and sensitivities are described in Chapter 15 (Ref. 3).

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

RCS LEAKAGE detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

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

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump level monitor, in combination with an N13/F18 gaseous activity monitor, provides an acceptable minimum. Containment sump level monitoring is performed by two redundant, seismically qualified level instruments. The LCO note clarifies that if leakage is prevented from draining to the sump, its level change measurements made by operable sump level instruments will not be valid for quantifying the leakage.

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

In MODE 5 or 6, the temperature is $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

(continued)

BASES

APPLICABILITY
(continued)

Containment sump level monitoring is a valid method for detecting leakage in MODES 1, 2, 3, and 4. The containment atmosphere N13/F18 radioactivity leakage measurement during MODE 1 is valid only for reactor power > 20% RTP. RCS inventory monitoring via the pressurizer level changes is valid in MODES 1, 2, 3, and 4 only when RCS conditions are stable, i.e., temperature is constant, pressure is constant, no makeup and no letdown.

The containment sump level change method of detecting leaks during MODES 1, 2, 3, and 4 is not valid while containment purge occurs or within 2 hours after the end of containment purge.

The containment atmosphere N13/F18 radioactivity leakage measurement during MODE 1 is not valid while containment purge occurs or within 2 hours after the end of containment purge.

The containment sump level change method of detecting leaks during MODES 1, 2, 3, and 4 is not valid during extremely cold outside ambient conditions when frost is forming on the interior of the containment vessel.

ACTIONS

A.1 and A.2

With two containment sump level channels inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere N13/F18 radioactivity monitor will provide indications of changes in LEAKAGE. Together with the atmosphere monitor, the periodic surveillance for RCS inventory balance, SR 3.4.8.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect LEAKAGE. A Note is added allowing that SR 3.4.8.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of one sump channel to OPERABLE status is required to regain the function in a Completion Time of 72 hours after the monitor's failure. This time is acceptable, considering the frequency and adequacy of the RCS inventory balance required by Action A.1.

(continued)

BASES

ACTIONS
(continued)

B.1.1, B.1.2, and B.2

With one gaseous N13/F18 containment atmosphere radioactivity-monitoring instrumentation channel inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or RCS inventory balanced, in accordance with SR 3.4.8.1, to provide alternate periodic information.

With a sample obtained and analyzed or an RCS inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the radioactivity monitor.

The 24 hours interval for grab samples or RCS inventory balance provides periodic information that is adequate to detect LEAKAGE. A Note is added allowing that SR 3.4.8.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, and makeup and letdown). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leak detection is available.

Required Action B.1 and Required Action B.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous N13/F18 containment atmosphere radioactivity monitor channel is inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.

C.1 and C.2

If a Required Action of Condition A or B cannot be met within the required Completion Time, the reactor 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 without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.10.1

SR 3.4.10.1 requires the performance of a CHANNEL CHECK of the containment atmosphere N13/F18 radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and risk and is reasonable for detecting off normal conditions.

SR 3.4.10.2

SR 3.4.10.2 requires the performance of a CHANNEL OPERATIONAL TEST (COT) on the atmosphere N13/F18 radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers risks and instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.10.3 and SR 3.4.10.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS Leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
 2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary LEAKAGE Detection Systems," U.S. Nuclear Regulatory Commission.
 3. Chapter 15, "Accident Analysis."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Specific Activity

BASES

BACKGROUND

The limits on RCS specific activity ensure that the doses due to postulated accidents are within the doses reported in Chapter 15.

The RCS specific activity LCO limits the allowable concentration of iodines and noble gases in the reactor coolant. The LCO limits are established to be consistent with a fuel defect level of 0.25 percent and to ensure that plant operation remains within the conditions assumed for shielding and Design Basis Accident (DBA) release analyses.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to limit the doses due to postulated accidents to within the values calculated in the radiological consequences analyses (as reported in Chapter 15).

APPLICABLE
SAFETY ANALYSES

The LCO limits on the reactor coolant specific activity are a factor in accident analyses that assume a release of primary coolant to the environment either directly as in a Steam Generator Tube Rupture (SGTR) or indirectly by way of leakage to the secondary coolant system and then to the environment (the Steam Line Break).

The events which incorporate the LCO values for primary coolant specific activity in the radiological consequence analysis include the following:

- Steam generator tube rupture (SGTR)
- Steam line break (SLB)
- Locked RCP rotor
- Rod ejection
- Small line break outside containment
- Loss of coolant accident (LOCA) (early stages)

The limiting event for release of primary coolant activity is the SLB. The SLB dose analysis considers the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

possibility of a pre-existing iodine spike (in which case the maximum LCO of 60 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 is assumed) as well as the more likely initiation of an iodine spike due to the reactor trip and depressurization. In the latter case, the LCO of 1.0 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 is assumed at the initiation of the accident, but the primary coolant specific activity is assumed to increase with time due to the elevated iodine appearance rate in the coolant. The reactor coolant noble gas specific activity for both cases is assumed to be the LCO of 280 $\mu\text{Ci/gm}$ DOSE EQUIVALENT XE-133. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 from LCO 3.7.4, "Secondary Specific Activity."

The LCO limits ensure that, in either case, the doses reported in Chapter 15 remain bounding.

The RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131, and the specific noble gas activity is limited to 280 $\mu\text{Ci/gm}$ DOSE EQUIVALENT XE-133. These limits ensure that the doses resulting from a DBA will be within the values reported in Chapter 15. Secondary coolant activities are addressed by LCO 3.7.4, "Secondary Specific Activity."

The SLB and SGTR accident analyses (Ref. 1 and 2) show that the offsite doses are within acceptance limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SLB or SGTR accident, lead to doses that exceed those reported Chapter 15.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq 500^\circ\text{F}$, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity are necessary to contain the potential consequences of a SGTR to within the calculated site boundary dose values.

For operation in MODE 3 with RCS average temperature $< 500^\circ\text{F}$ and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to verify that DOSE EQUIVALENT I-131 is $\leq 60 \mu\text{Ci/gm}$. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to normal within 48 hours. If the concentration cannot be restored to within the LCO limit in 48 hours, it is assumed that the LCO violation is not the result of normal iodine spiking.

A Note to the Required Action 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.

B.1 and B.2

With DOSE EQUIVALENT XE-133 in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The allowed Completion Time of 4 hours is required to obtain and analyze a sample.

The change to MODE 3 and RCS average temperature $< 500^\circ\text{F}$ lowers the saturation pressure of the reactor coolant below the set points of the main steam safety valves, and prevents venting the SG to the environment in a SGTR event. 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, without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is $> 60 \mu\text{Ci/gm}$., the reactor must be brought to MODE 3 with

(continued)

BASES

ACTIONS

C.1 (continued)

RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operation experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.11.1

SR 3.4.11.1 requires performing a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This is a quantitative measure of radionuclides with half lives longer than 15 minutes. This Surveillance provides an indication of any increase in the release of noble gas activity from fuel rods containing cladding defects.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the unlikelihood of a significant increase in fuel defect level during the time.

SR 3.4.11.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when increased releases of iodine from the fuel (iodine spiking) is apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level. The Frequency, between 2 and 6 hours after a power change of $\geq 15\%$ RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failures; samples at other times would provide inaccurate results.

REFERENCES

1. Section 15.1.5, "Steam System Piping Failure."
 2. Section 15.6.3, "Steam Generator Tube Rupture."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Automatic Depressurization System (ADS) - Operating

BASES

BACKGROUND

The ADS is designed to assure that core cooling and injection can be achieved for Design Basis Accidents (DBA). The four stages of ADS valves are sequenced in coordination with the passive core cooling system injection performance characteristics.

The ADS consists of 10 flow paths arranged in four different stages that open sequentially (Ref. 1). Stages 1, 2, and 3 each include 2 flow paths. Each of the stage 1, 2, 3 flow paths has a common inlet header connected to the top of the pressurizer. The outlets of the stage 1, 2, 3 flow paths combine into one of the two common discharge lines to the spargers located in the incontainment refueling water storage tank (IRWST). The first stage valves are 4 inch valves with DC motor operators. The second and third stage valves are 8 inch valves with DC motor operators. An OPERABLE stage 1, 2, or 3 automatic depressurization flow path consists of two OPERABLE normally closed motor operated valves, in series.

Stage 4 includes 4 flow paths. The fourth stage ADS valves are squib valves. The four fourth stage flow paths connect directly to the top of the reactor coolant hot legs and vent directly into the associated steam generator compartment. An OPERABLE stage 4 flow path consists of an open motor operated valve and an OPERABLE closed squib valve. These motor operated valves are not required to be OPERABLE because they are open.

The automatic depressurization valves are designed to open automatically when actuated, and to remain open for the duration of any automatic depressurization event. The valves are actuated sequentially. The stage 1 valves are actuated on a low core makeup tank (CMT) level. Stages 2 and 3 are actuated on the stage 1 signal plus time delays. Stage 4 is actuated on a Low 2 CMT level signal with a minimum time delay after stage 3. Stage 4 is blocked from actuating at normal RCS pressure.

In order to perform a controlled, manual depressurization of the RCS, the valves are opened starting with the first

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BASES

BACKGROUND
(continued)

stage. The first stage valves can also be modulated to perform a partial RCS depressurization if required. ADS stage 1, 2, 3 valves may be manually operated under controlled conditions for testing purposes.

ADS stages 1, 2 and 3 valves are designed to open relatively slowly, from approximately 25 seconds for the first stage valves, to approximately 70 seconds for the second and third stage valves.

The ADS valves are powered by batteries. In the unlikely event that offsite and onsite AC power is lost for an extended period of time, a timer will actuate ADS within 24 hours of the time at which AC power is lost, before battery power has been degraded to the point where the valves cannot be opened.

The number and capacity of the ADS flow paths are selected so that adequate safety injection is provided from the accumulators, IRWST and containment recirculation for the limiting DBA loss of coolant accident (LOCA). For small break LOCAs the limiting single failure is the loss of one fourth stage flow path (Ref. 2). The PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of two (or more) flow paths. The ADS PRA success criteria following a LOCA or non-LOCA with failure of other decay heat removal features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1, 2, 3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation.

APPLICABLE
SAFETY ANALYSES

For non-LOCA events, use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the CMTs may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation.

For events which involve a loss of primary coolant inventory, such as a LOCA, the ADS will be actuated, allowing for injection from the accumulators, the IRWST, and the containment recirculation (Ref. 2).

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

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The requirement that the 16 ADS valves be OPERABLE ensures that upon actuation, the depressurization of the RCS will proceed smoothly and completely, as assumed in the DBA safety analyses.

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BASES

LCO
(continued)

For the ADS to be considered OPERABLE, the 16 ADS valves must be closed and OPERABLE (capable of opening on an actuation signal). In addition, the stage 4 motor operated isolation valves must be open. These stage 4 motor operated isolation valves are not required to be OPERABLE because they are maintained open per SR 3.4.12.1.

APPLICABILITY

In MODES 1, 2, 3 and 4 the ADS must be OPERABLE to mitigate the potential consequences of any event which causes a reduction in the RCS inventory, such as a LOCA.

The requirements for the ADS in MODES 5 and 6 are specified in LCO 3.4.13, "Automatic Depressurization System (ADS) - Shutdown, RCS Intact," and LCO 3.4.14, "Automatic Depressurization System - Shutdown, RCS Open."

ACTIONS

A.1

If any one, or if two flow paths, consisting of one stage 1 and one stage 2 or 3, are determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function as long as a single failure does not also occur. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is reasonable based on the capability of the remaining ADS valves to perform the required safety functions assumed in the safety analyses and the low probability of a DBA during this time period. This Completion Time is the same as is used for two train ECCS systems which are capable of performing their safety function without a single failure.

B.1 and B.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.12 are not met for reasons other than Condition A, the plant must be brought to MODE 5 where the probability and consequences on an event are minimized. 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, without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.12.1

Each stage 4 ADS isolation motor operated valve must be verified to be open every 12 hours. Note that these valves receive confirmatory open signals. The Surveillance Frequency is acceptable considering valve position is manually monitored in the control room.

SR 3.4.12.2

This Surveillance requires verification that each ADS stage 1, 2, 3 valve strokes to its fully open position. Note that this surveillance is performed during shutdown conditions.

The Surveillance Frequency for demonstrating valve OPERABILITY references the Inservice Testing Program.

SR 3.4.12.3

This Surveillance requires verification that each ADS stage 4 squib valve is OPERABLE in accordance with the Inservice Testing Program. The Surveillance Frequency for verifying valve OPERABILITY references the Inservice Testing Program.

The squib valves will be tested in accordance with ASME Section XI which specifies valve testing in accordance with the ASME OM Code. The applicable ASME OM Code squib valve requirements are specified in paragraph 4.6, Inservice Tests for Category D Explosively Actuated Valves. The requirements include actuation of a sample of the installed valves each 2 years and periodic replacement of charges.

REFERENCES

1. Section 6.3, "Passive Core Cooling System."
 2. Section 15.6, "Decrease in Reactor Coolant Inventory."
 3. AP1000 Probabilistic Risk Assessment, Appendix A.
 4. Section 3.9.6, "Inservice Testing of Pumps and Valves."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 Automatic Depressurization System (ADS) - Shutdown, RCS Intact

BASES

BACKGROUND A description of the ADS is provided in the Bases for LCO 3.4.12, "Automatic Depressurization System (ADS) - Operating."

APPLICABLE SAFETY ANALYSES For postulated events in MODE 5 with the RCS pressure boundary intact, the primary protection is the Passive Residual Heat Removal Heat Exchanger (PRHR HX). Use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the core makeup tanks (CMTs) may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation.

No LOCAs are postulated during plant operation in MODE 5, however loss of primary coolant through leakage or inadvertent draining may occur. For such shutdown events occurring in MODE 5 it is anticipated that the ADS will be actuated, allowing injection from the in-containment refueling water storage tank (IRWST) and the containment recirculation if containment flooding occurs (Ref. 2).

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

LCO The requirement that 9 ADS flow paths be OPERABLE assures that upon actuation, the depressurization of the RCS will proceed smoothly and completely, as assumed in the DBA safety analyses.

An ADS stage 1, 2, or 3 flow path is considered OPERABLE if both valves in the line are closed and OPERABLE (capable of opening on an actuation signal). In addition, an ADS stage 4 flow path is operable if the motor operated isolation valve is open and the squib valve is closed and OPERABLE (capable of opening on an actuation signal).

APPLICABILITY In MODE 5 with the reactor coolant pressure boundary (RCPB) intact, 9 flow paths of the ADS must be OPERABLE to mitigate the potential consequences of any event which causes a reduction in the RCS inventory, such as a LOCA.

(continued)

BASES

APPLICABILITY
(continued)

The requirements for the ADS in MODES 1 through 4 are specified in LCO 3.4.12, "Automatic Depressurization System (ADS) – Operating;" and in MODE 5 with the RCS pressure boundary open and MODE 6 in LCO 3.4.14, "Automatic Depressurization System (ADS) – Shutdown, RCS Open."

ACTIONS

A.1

If any one, or if two flow paths, consisting of one stage 1 and one stage 2 or 3, are determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is acceptable since the OPERABLE ADS paths can mitigate shutdown events without a single failure.

B.1

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.13 are not met for reasons other than Condition A, the plant must be placed in a MODE in which this LCO does not apply. Action must be initiated, immediately, to place the plant in MODE 5 with the RCS pressure boundary open and $\geq 20\%$ pressurizer level.

SURVEILLANCE
REQUIREMENTS

SR 3.4.13.1

The LCO 3.4.12 Surveillance Requirements (SR 3.4.12.1) are applicable to the ADS valves required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.4.12 for a discussion of each SR.

REFERENCES

1. AP1000 Probabilistic Risk Assessment, Appendix A.
 2. Section 19E.4, "Safety Analyses and Evaluations."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 Automatic Depressurization System (ADS) - Shutdown, RCS Open

BASES

BACKGROUND A description of the ADS is provided in the Bases for LCO 3.4.12, "Automatic Depressurization System (ADS) - Operating."

APPLICABLE SAFETY ANALYSES When the plant is shutdown with the RCS depressurized, the core makeup tanks (CMTs) are isolated to prevent CMT injection. Since the ADS is actuated by low CMT level, automatic actuation of the ADS is not available. The required ADS stage 1, 2, and 3 vent paths are opened and two ADS stage 4 flow paths are OPERABLE to ensure that in-containment refueling water storage tank (IRWST) injection and containment recirculation can occur, if needed to mitigate events requiring RCS makeup, boration or core cooling (Ref. 1).

The ADS vent path must be maintained until the upper internals are removed, providing an adequate vent path for IRWST injection.

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

LCO The requirement that ADS stage 1, 2, and 3 flow paths be open, from the pressurizer through the spargers into the IRWST, and that two ADS stage 4 flow paths be OPERABLE assures that sufficient vent area is available to support IRWST injection.

The Note allows closure of the RCS pressure boundary when the pressurizer level is < 20% to facilitate vacuum refill following mid-loop operations to establish a pressurizer water level ≥ 20%. Prior to closure of the ADS valves, compliance with LCO 3.4.13, ADS - Shutdown, RCS Intact, should be verified.

APPLICABILITY In MODE 5 with the reactor coolant system pressure boundary (RCPB) open or pressurizer level < 20% and in MODE 6 with the upper internals in place, the stage 1, 2, and 3 ADS vent paths must be open and two ADS stage 4 flow paths be OPERABLE.

(continued)

BASES

APPLICABILITY
(continued)

The requirements for the ADS in MODES 1 through 4 are specified in LCO 3.4.12, "Automatic Depressurization System (ADS) – Operating;" and in MODE 5 with the RCPB intact in LCO 3.4.13, "Automatic Depressurization System (ADS) – Shutdown, RCS Intact."

ACTIONS

A.1 and A.2

If one required ADS stage 1, 2, or 3 flow path is closed, action must be taken to open the affected path or establish an alternative flow path within 72 hours. In this Condition the remaining open ADS stage 1, 2, and 3 flow paths and the OPERABLE ADS stage 4 flow paths are adequate to perform the required safety function without an additional single failure. The stage 4 valves would have to be opened by the operator in case of an event in this MODE. The required vent area may be restored by opening the affected ADS flow path or an alternate vent path with an equivalent area. Considering that the required function is available in this Condition a Completion Time of 72 hours is acceptable.

B.1 and B.2

If one required ADS stage 4 flow path is closed and inoperable, action must be taken to establish an alternative flow path, or restore at least two stage 4 flow paths to OPERABLE status within 36 hours. In this Condition the remaining open ADS stage 1, 2, and 3 flow paths and the one OPERABLE ADS stage 4 flow path are adequate to perform the required safety function without an additional single failure. The required vent area may be restored by opening an alternate vent path with an equivalent area. Alternatively, two stage 4 flow paths may be restored to OPERABLE status. Therefore a Completion Time of 36 hours is considered acceptable.

C.1 and C.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.14 are not met for reasons other than Conditions A or B while in MODE 5, the plant must be placed in a condition which minimizes the potential for requiring ADS venting and IRWST injection. The time to RCS boiling is maximized by increasing RCS inventory to $\geq 20\%$ pressurizer level and maintaining RCS temperature as low as practical.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Additionally, action to suspend positive reactivity additions is required to ensure that the shutdown margin is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

D.1 and D.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.14 are not met for reasons other than Conditions A or B while in MODE 6, the plant must be placed in a condition which precludes the need for the ADS vent paths. Action must be initiated, immediately, to remove the upper internals, providing the required vent path. The time to RCS boiling is maximized by increasing RCS inventory and maintaining RCS temperature as low as practical. Additionally, action to suspend positive reactivity additions is required to ensure that the shutdown margin is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

SURVEILLANCE
REQUIREMENTS

SR 3.4.14.1

Each required ADS flow path is verified to be open by verifying that the stage 1, 2, and 3 valves are in their fully open position every 12 hours, as indicated in the control room. This Surveillance Frequency is acceptable based on administrative controls which preclude repositioning the valves.

SR 3.4.14.2

The LCO 3.4.12 Surveillance Requirements (SR 3.4.12.1 and SR 3.4.12.3) are applicable to the stage 4 ADS valves required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.4.12 for a discussion of each SR.

REFERENCES

1. Section 19E.4, "Safety Analyses and Evaluations."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System limits RCS pressure at low temperatures so that the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the limits which set the maximum allowable setpoints for the Normal Residual Heat Removal System (RNS) suction relief valve. LCO 3.4.3 provides the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

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

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

This LCO provides RCS overpressure protection by having a maximum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires isolating the accumulators. The pressure relief capacity requires the RNS suction relief valve or a depressurized RCS and an RCS vent of sufficient size. The RNS suction relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

(continued)

BASES

BACKGROUND
(continued)

RNS Suction Relief Valve Requirements

During the LTOP MODES, the RNS system is operated for decay heat removal. Therefore, the RNS suction isolation valves are open in the piping from the RCS hot legs to the inlet of the RNS system. While these valves are open, the RNS suction relief valve is exposed to the RCS and able to relieve pressure transients in the RCS.

The RNS suction relief valve is a spring loaded, water relief valve with a pressure tolerance and an accumulation limit established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.

The RNS suction isolation valves must be open to make the RNS suction relief valves OPERABLE for RCS overpressure mitigation.

RCS Vent Requirements

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

For an RCS vent to meet the flow capacity requirement, it may require removing one or more pressurizer safety valves or manually opening one or more Automatic Depressurization System (ADS) valves. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE
SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with the RCS temperature above 275°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. When the RNS is aligned and open to the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

RCS, overpressure protection is provided by the RNS suction relief valve, or a depressurized RCS and a sufficiently sized open RCS vent.

The actual temperature at which the pressure in the P/T limit curve falls below the suction relief setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RNS suction relief valve, or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients. The events listed below were used in the analysis to size the RNS suction relief valve. Therefore, any events with a mass or heat input greater than the listed events cannot be accommodated and must be prevented.

Mass Input

- a. Makeup water flow rate to the RCS assuming both CVS makeup pumps are in operation and letdown is isolated.

Heat Input

- a. Restart of one reactor coolant pump (RCP) with water in the steam generator secondary side 50°F hotter than the primary side water, and the RCS water solid.

RNS Suction Relief Valve Performance

Since the RNS suction relief valve does not have a variable P/T lift setpoint, the analysis must show that with chosen setpoint, the relief valve will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the minimum of either the P/T limit curve or 110 percent of the design pressure of the normal residual heat removal system. The current analysis shows that up to a temperature of 100°F, the mass input transient is limiting, and above this temperature the heat input transient is limiting.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

To prevent the possibility of a heat input transient, and thereby limit the required flow rate of the RNS suction relief valve, an administrative requirement has been imposed that does not allow an RCP to be started with the pressurizer water level above 92% and the RCS temperature above 200°F. Under these imposed conditions, the transient created by the startup of an RCP when the RCS temperature is above 200°F can be accommodated without additional pressure relief.

RCS Vent Performance

With the RCS depressurized, a vent size of [5.4] square inches is capable of mitigating a limiting overpressure transient. The area of the vent is equivalent to the area of the inlet pipe to the RNS suction relief valve so the capacity of the vent is greater than the flow possible with either the mass or heat input transient, while maintaining the RCS pressure less than the minimum of either the maximum pressure on the P/T limit curve or 110 percent of the design pressure of the normal residual heat removal system.

The required vent area may be obtained by opening one ADS flow path.

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

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

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

LCO

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

To limit the coolant input capability, the LCO requires all accumulator discharge isolation valves closed and immobilized, when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed in the PTLR.

(continued)

BASES

LCO
(continued)

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

- a. One OPERABLE RNS suction relief valve; or

An RNS suction relief valve is OPERABLE for LTOP when both RNS suction isolation valves in one flow path are open, its setpoint is within limits, and testing has proven its ability to open at this setpoint.

- b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of $\geq [5.4]$ square inches.

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

APPLICABILITY

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

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.7, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 with the RNS isolated or RCS temperature $\geq 275^\circ\text{F}$.

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

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves.

This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

(continued)

BASES (continued)

ACTIONS

A.1, B.1, and B.2

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

If isolation is needed and cannot be accomplished in 1 hour, Required Action B.1 and Required Action B.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 275°F, the accumulator pressure cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

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

C.1 and C.2

If the RNS suction relief valve is inoperable and the RCS is not depressurized, there is a potential to overpressurize the RCS and exceed the limits allowed in LCO 3.4.3. The suction relief valve is considered inoperable if the RNS isolation valves have isolated the RNS from the RCS in such a way that the suction relief valve cannot perform its intended safety function, or if the valve itself will not operate to perform its intended safety function.

Under these conditions, Required Actions C.1 or C.2 provide two options, either of which must be accomplished in 12 hours. If the RNS suction relief valve cannot be restored to operable status, the RCS must be depressurized and vented with a RCS vent which provides a flow area sufficient to mitigate any of the design low temperature overpressure events.

The 12 hour Completion Time represents a reasonable time to repair the relief valve, open the RNS isolation valves or otherwise restore the system to OPERABLE status, or depressurize and vent the RCS, without imposing a lengthy period when the LTOP system is not able to mitigate a low temperature overpressure event.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.15.1

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, the accumulator discharge isolation valves are verified closed and locked out. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the main control room to verify the required status of the equipment.

SR 3.4.15.2

The RNS suction relief valve shall be demonstrated OPERABLE by verifying two RNS suction isolation valves in one flow path are open. This Surveillance is only performed if the RNS suction relief valve is being used to satisfy this LCO.

The RNS suction isolation valves are verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RNS suction isolation valves remain open.

SR 3.4.15.3

The RCS vent of $\geq [5.4]$ square inches is proven OPERABLE by verifying its open condition either:

- a. Once every 12 hours for a valve that is not locked (valves that are sealed or secured in the open position are considered "locked" in this context) or
- b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position or a removed pressurizer safety valve or open manway also fits this category).

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.15b.

SR 3.4.15.4

The RNS suction relief valve shall be demonstrated OPERABLE by verifying that two RNS suction isolation valves in one flow path are open and by testing it in accordance with the

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.15.4 (continued)

Inservice Testing Program. (Refer to SR 3.4.15.2 for the RNS suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RNS suction relief valve is being used to meet this LCO. The ASME Code, Section XI (Ref. 5), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."
 2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operation."
 3. ASME Boiler and Pressure Vessel Code, Section III.
 4. Section 5.2.2, "Overpressure Protection."
 5. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Pressure Isolation Valve (PIV) Integrity

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define the RCS pressure boundary as all those pressure containing components such as pressure vessels, piping, pumps, and valves which are connected to the reactor coolant system, up to and including the outermost containment isolation valve in system piping which penetrates primary reactor containment, the second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment, and the reactor coolant system safety and relief valves. This includes any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can experience varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The AP1000 PIVs are listed in Chapter 3, Table 3.9-18. The RCS PIV Leakage LCO allows RCS high pressure operation when PIV leakage has been verified.

The purpose of this specification is to prevent overpressure failure or degradation of low pressure portions of connecting systems. The following criteria was used in identifying PIVs for inclusion in the specification. A valve was included in this specification if it's failure may result in:

1. Failure of low pressure portions of connected systems, such as a Loss of Coolant Accident (LOCA) outside of containment, which could place the plant in an unanalyzed condition.

(continued)

BASES

BACKGROUND
(continued)

2. Degradation of low pressure portions of connected systems, such as damage to a core cooling system, which could degrade a safety related function that mitigates a DBA.

Valves considered for inclusion in this specification are used to isolate the RCS from the following connected systems:

- a. Passive Core Cooling System (PXS) Accumulators;
- b. Normal Residual Heat Removal System (RNS); and
- c. Chemical and Volume Control System (CVS).

The RNS pressure boundary isolation valves are considered to meet the first criterion for inclusion in this specification. The PXS accumulator check valves were determined to meet the second PIV criteria for inclusion in this specification. It is determined that the CVS PIVs do not meet either criteria for inclusion in this specification.

The PIVs that are addressed by this specification are listed in Chapter 3, Table 3.9-18.

The CVS pressure isolation valves were not included in this specification based on the defined criteria. The justification for excluding the CVS PIVs is discussed in the following paragraph.

The CVS contains four high pressure / low pressure connections with the RCS. Since the portion of the CVS which is located inside reactor containment is designed to full RCS pressure, the high pressure / low pressure interfaces with the RCS are the lines that penetrate the reactor containment. The CVS lines that penetrate containment include the makeup line, the letdown line to the

(continued)

BASES

BACKGROUND
(continued)

Liquid Radwaste System, the hydrogen supply line, and the demineralizer resin sluice line used to transfer spent resins from the demineralizers to the Solid Radwaste System. These lines each contain two safety related containment isolation valves which are addressed by the Containment Isolation Specification (LCO 3.6.3). In addition to the containment isolation valves in each of the CVS lines that interface with the RCS, there are additional valves in each line that provide diverse isolation capability. Since more restrictive requirements are imposed by LCO 3.6.3, the CVS isolation valves are not included in this LCO.

Since the purpose of this LCO is to verify that the PIVs have not suffered gross failures, the valve leakage test in conjunction with tests specified in the IST program provide an acceptable method of determining valve integrity. The ability of the valves to transition from open to closed provides assurance that the valve can perform its pressure isolation function as required. A small amount leakage through these valves is allowed, provided that the integrity of the valve was demonstrated.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system or the failure of a safety related function to mitigate a DBA.

APPLICABLE
SAFETY ANALYSES

Pressure isolation valve integrity is not considered in any design basis accident analyses. This specification provides for monitoring the condition of the reactor coolant pressure boundary to detect degradation which could lead to accidents or which could impair a connected system's ability to mitigate DBAs.

RCS PIV integrity satisfies, Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually small. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per inch nominal valve size up to a maximum of 5 gpm per valve. This limit is well within the makeup capability of the CVS makeup pumps. This leak rate will not result in the overpressure of a connected low pressure system. Reference 5 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage of the valve. In such cases, the observed leakage rate at lower differential pressures can be assumed to be the leakage at the maximum pressure differential. Verification that the valve leakage diminishes with increasing pressure differential is sufficient to verify that the valve characteristics are such that higher service pressure results in a decrease in overall leakage.

APPLICABILITY In MODES 1, 2, and 3 and MODE 4, with RCS not being cooled by the RNS, this LCO applies when the RCS is pressurized.

In MODE 4, with RNS in operation, and MODES 5 and 6, the RCS pressure is reduced and is not sufficient to overpressurize the connected low pressure systems.

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

(continued)

BASES

ACTIONS
(continued)

A.1

With one or more PIVs inoperable, the affected flow path(s) must be isolated. Required Action A.1 is modified by a Note that the valves used for isolation must meet the same integrity requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Required Action A.1 requires that the isolation with one valve must be performed within 8 hours. Eight hours provides time to verify IST compliance for the alternate isolation valve and isolate the flow path. The 8 hour Completion Time allows the actions and restricts the operation with inoperable isolation valves.

A.2

Required Action A.2 specifies that a second OPERABLE PIV can be shown to meet the leakage limits within 8 hours. This valve is required to be a check valve, or a closed valve, if it isolates a line that penetrates containment. For the accumulator valves, the normally open accumulator isolation valve is a suitable replacement PIV, but can remain open because leakage into the accumulator is continuously monitored. If leakage into the accumulators increased to the allowable operational leakage limit, then the valve could be used to isolate the accumulators from the RCS.

The 72 hour Completion Time allows the actions and restricts the operation with inoperable isolation valves.

B.1 and B.2

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

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.16.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch nominal valve size up to a minimum of 5 gpm 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 shall be performed every 24 months, a typical refueling cycle. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 4) as contained in the Inservice Testing Program and is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 5).

REFERENCES

1. 10 CFR 50.2.
 2. 10 CFR 50.55a(c).
 3. 10 CFR 50, Appendix A, Section V, GDC 55.
 4. 10 CFR 50.55a(g).
 5. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Reactor Vessel Head Vent (RVHV)

BASES

BACKGROUND

The reactor vessel head vent (RVHV) is designed to assure that long-term operation of the Core Makeup Tanks (CMTs) does not result in overfilling of the pressurizer during Condition II Design Basis Accidents (DBAs). The RVHV can be manually actuated by the operators in the main control room to reduce the pressurizer water level during long-term operation of the CMTs.

The RVHV consists of two parallel flow paths each containing two RVHV isolation valves in series. The RVHV valves are connected to the reactor vessel head via a common line. The outlets of the RVHV flow paths combine into one common discharge line which connects to a single ADS discharge header that discharges to spargers located in the incontainment refueling water storage tank (IRWST). The RVHV valves are 1 inch valves with DC solenoid operators.

The RVHV valves are designed to open when actuated by the operator, and to reclose when actuated by the operator from the main control room.

The number and capacity of the RVHV flow paths are selected so that letdown flow from the RCS is sufficient to prevent pressurizer overfill for events where extended operation of the CMTs causes the pressurizer water level to increase. Although realistic evaluations of the Condition II non-LOCA events does not result in pressurizer overfill, conservative analyses of some of these events can result in pressurizer overfill if no operator actions are assumed.

APPLICABLE
SAFETY ANALYSES

For Condition II non-LOCA events, such as inadvertent passive core cooling system operation and chemical and volume control system malfunction, the use of the RVHV may be required to prevent long-term pressurizer overfill (Reference 1).

For LOCA events, the RVHV is not required.

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

(continued)

BASES (continued)

LCO The requirement that all four RVHV valves be OPERABLE ensures that upon actuation, the RVHV can reduce the pressurizer water level as assumed in the DBA safety analyses.

For the RVHV to be considered OPERABLE, all four valves must be closed and OPERABLE (capable of opening from the main control room).

APPLICABILITY In MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS, the RVHV must be OPERABLE to mitigate the potential consequences of any event which causes an increase in the pressurizer water level that could otherwise result in overfilling of the pressurizer.

In MODE 4, with the RCS being cooled by the RNS, and in MODES 5 and 6, operation of the CMTs or CVS will not result in a pressurizer overflow event.

ACTIONS A.1

If one or two RVHV valves in a single flow path are determined to be inoperable, the flow path is inoperable. The remaining OPERABLE RVHV flow path is adequate to perform the required safety function. A Completion Time of 72 hours is acceptable since the OPERABLE RVHV paths can mitigate DBAs without a single failure.

B.1

If both flow paths are determined to be inoperable, the RVHV is degraded such that the system is not available for some DBA non-LOCA analysis for which may be required. A Completion Time of 6 hours is permitted to restore at least one flow path. This Completion Time is acceptable considering that the realistic analysis of these non-LOCA events do not result in pressurizer overflow.

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.17 are not met for reasons other than Conditions A or B, the plant must be brought to MODE 4 where the probability and consequences of an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.17.1

The dedicated component level remote manual valve switches in the main control room shall be used to stroke each RVHV valve to demonstrate OPERABILITY of the controls.

This Surveillance requires verification that each RVHV valve strokes to its fully open position. The Surveillance Frequency for demonstrating valve OPERABILITY references the Inservice Testing Program.

REFERENCES

1. Section 15.5, "Increase in Reactor Coolant System Inventory."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 Chemical and Volume Control System (CVS) Makeup Isolation Valves

BASES

BACKGROUND

One of the principle functions of the CVS system is to maintain the reactor coolant inventory by providing water makeup for reactor coolant system (RCS) leakage, shrinkage of the reactor coolant during cooldowns, and RCS boron concentration changes. In the automatic makeup mode of operation, the pressurizer water level starts and stops CVS makeup to the RCS.

Although the CVS is not considered a safety related system, certain isolation functions of the system are considered safety related functions. The appropriate isolation valves have been classified and designed as safety related. One of the safety related functions provided by the CVS is the termination of RCS makeup to prevent overfilling of the pressurizer during non-LOCA transients or to prevent steam generator overfilling during a steam generator tube rupture. The CVS makeup line containment isolation valves provide this RCS makeup isolation function.

APPLICABLE
SAFETY ANALYSES

One of the initial assumptions in the analysis of several non-LOCA events and during a steam generator tube rupture accident is that excessive CVS makeup to the RCS may aggravate the consequences of the accident. The need to isolate the CVS makeup to the RCS is detected by the pressurizer level instruments or the steam generator narrow range level instruments. These instruments will supply a signal to the makeup line containment isolation valves in the CVS causing these valves to close and terminate RCS makeup. Thus the CVS makeup isolation valves are components which function to mitigate an accident.

CVS isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO The requirement that at least two CVS makeup isolation valves be OPERABLE assures that there will be redundant means available to terminate CVS makeup to the RCS during a non-LOCA event or a steam generator tube rupture accident should that become necessary.

APPLICABILITY The requirement that at least two CVS makeup isolation valves be OPERABLE is applicable in MODES 1, 2, 3, and 4 with the normal residual heat removal system (RNS) suction to the RCS not open because a pressurizer overfill event or steam generator tube rupture accident is considered possible in these MODES, and the automatic closure of these valves is assumed in the safety analysis.

In the applicable MODES, the need to isolate the CVS makeup to the RCS is detected by the pressurizer level instruments (high 1 setpoint coincident with safeguards actuation or high 2 setpoint) or the steam generator narrow range level instruments (high 2 setpoint).

This isolation function is not required in MODE 4 with the RNS suction open to the RCS or in lower MODES. In such MODES, pressurizer or steam generator overfill is prevented by the RNS suction relief valve.

ACTIONS

A.1

If only one CVS makeup isolation valve is OPERABLE, the second valve must be restored to OPERABLE status in 72 hours. The allowed Completion Time assures expeditious action will be taken, and is acceptable because the safety function of automatically isolating RCS makeup can be accomplished by the redundant isolation valve.

B.1

If the Required Actions and associated Completion Time of Condition A are not met, or if both CVS makeup isolation valves are not OPERABLE (i.e., not able to be closed automatically), then the makeup flow path to the RCS must be isolated. Isolation can be accomplished by manually closing

(continued)

BASES

ACTIONS

B.1 (continued)

the CVS makeup isolation MOVs or alternatively, manual valve(s) in the makeup line between the makeup pumps and the RCS.

The Action is modified by a Note allowing the flow path 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 main control room. In this way, the flow path can be rapidly isolated when a need for isolation is indicated.

SURVEILLANCE
REQUIREMENTS

SR 3.4.18.1

Verification that the RCS makeup isolation valves are OPERABLE, by stroking each valve closed, demonstrates that the valves can perform their safety related function. The Frequency is in accordance with the Inservice Testing Program.

SR 3.4.18.2

Verification that the RCS makeup isolation valves closure times are less than that assumed in the safety analysis, is performed by measuring the time required for each valve to close. The Frequency is in accordance with the Inservice Testing Program.

REFERENCES

1. Chapter 15, "Accident Analysis."
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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the PXS accumulators are to supply water to the reactor vessel during the refill phase of a large break Loss Of Coolant Accident (LOCA), to provide Reactor Coolant System (RCS) makeup for a small break LOCA, and to provide Reactor Coolant System (RCS) boration for steam line breaks (Ref. 2).

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

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The accumulator inventory is available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core.

The accumulators are pressure vessels, partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required for them to perform their function. Internal accumulator pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the static accumulator pressure.

Each accumulator is piped into the reactor vessel via an accumulator line and is isolated from the RCS by two check valves in series.

A normally open motor operated valve is arranged in series with the check valves. Upon initiation of a safeguards actuation signal, the normally open valves receive a confirmatory open signal.

(continued)

BASES

BACKGROUND
(continued)

Power lockout and position alarms ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 603-1991 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without being subject to a single failure.

The accumulator size, water volume, and nitrogen cover pressure are selected so that both of the accumulators are sufficient to recover the core cooling before significant clad melting or zirconium water reaction can occur following a large break LOCA. One accumulator is adequate during a small break LOCA where the entire contents of one accumulator can possibly be lost via the pipe break. This accumulator performance is based on design basis accident (DBA) assumptions and models (Ref. 3). The Probabilistic Risk Assessment (PRA) (Ref. 4) shows that one of the two accumulators is sufficient for a large break LOCA caused by spurious ADS actuation and that none of the accumulators are required for small break LOCAs, assuming that at least one core makeup tank (CMT) is available. In addition, both accumulators are required for a large break LOCA caused by the break of a cold leg pipe; the probability of this break has been significantly reduced by incorporation of leak-before-break.

APPLICABLE
SAFETY ANALYSES

The accumulators are assumed to be OPERABLE in both the large and small break LOCA analyses at full power (Ref. 3) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

For a small break LOCA, a large range of break sizes and locations were analyzed to verify the adequacy of the design. The cases analyzed include the rupture of one 8 inch direct vessel injection line and several smaller break sizes. Acceptable PXS performance was demonstrated.

For a larger LOCA, including a double ended RCS piping rupture, the PXS can provide a sufficiently large flow rate, assuming both accumulators are OPERABLE, to quickly fill the reactor vessel lower plenum and downcomer. Both

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

accumulators, in conjunction with the CMTs, ensure rapid reflooding of the core. For a large LOCA, both lines are available since an 8 inch line break would be a small LOCA.

Following a non-LOCA event such as a steamline break, the RCS experiences a decrease in temperature and pressure due to an increase in energy removal by the secondary system. The cooldown results in a reduction of the core SHUTDOWN MARGIN with a potential for return to power. During such an event the accumulators provide injection of borated water to assist the CMT's boration to mitigate the reactivity transient and ensure the core remains shut down.

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

LCO

This LCO establishes the minimum conditions necessary to ensure that sufficient accumulator flow will be available, assuming the minimum requirements of all other PXS components are met, to meet the necessary acceptance criteria established for core cooling by 10 CFR 50.46 (Ref. 5). These conditions are:

- a. Maximum fuel element cladding temperature is $\leq 2200^{\circ}\text{F}$;
- 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 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 accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For an accumulator to be OPERABLE, the isolation valve must be fully open with power removed, and the limits established in the Surveillance Requirements for contained water, boron concentration, and nitrogen cover pressure must be met.

(continued)

BASES (continued)

APPLICABILITY In MODES 1 and 2, and in MODES 3 and 4 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures \leq 1000 psig, the rate of RCS blowdown is such that adequate injection flow from other sources exists to retain peak clad temperatures below the 10 CFR 50.46 limit of 2200°F.

In MODES 3 and 4 with RCS pressure \leq 1000 psig, and in MODES 5 and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows the RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, action must be taken to restore the parameter.

Deviations in boron concentration are expected to be slight, considering that the pressure and volume are verified once per 24 hours. For one accumulator, boron concentration not within limits will have an insignificant effect on the ability of the accumulators to perform their safety function. Therefore, a Completion Time of 72 hours is considered to be acceptable.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 8 hours. With one accumulator inoperable, the remaining accumulator is capable of providing the required safety function, except for one low probability event (large cold leg LOCA) discussed in the background section. The effectiveness of one accumulator is demonstrated in analysis performed to justify PRA success criteria (Ref. 4). The analysis contained in this reference shows that for a range

(continued)

BASES

ACTIONS

B.1 (continued)

of other events including small LOCAs and large hot leg LOCAs that with one accumulator unavailable the core is adequately cooled. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk (Ref. 7).

The 8 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the Required Action and associated Completion Time of Conditions A or B are not met, the plant must be placed in a MODE or condition in which the LCO does not apply. This is done by placing the plant in MODE 3 within 6 hours and with pressurizer pressure to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If more than one accumulator 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

Each accumulator valve should be verified to be fully open every 12 hours. This verification ensures each accumulator isolation valve is fully open, as indicated in the control room, and timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a partially closed valve could result in not meeting DBA analyses assumptions (Ref. 3). A 12 hour Frequency is considered reasonable in view of the other administrative controls which ensure that a mispositioned isolation valve is unlikely.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.1.2 and 3.5.1.3

Verification every 12 hours of the borated water volume and nitrogen cover pressure in each accumulator is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Considering that control room alarms are provided for both parameters these limits are effectively subject to continuous monitoring. The 12 hour Frequency is considered reasonable considering the availability of the control room alarms and the likelihood that, with any deviation which may occur, the accumulators will perform their safety function with slight deviations in these parameters.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days, since the static design of the accumulators 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 in-leakage. Sampling the affected accumulator within 6 hours after a 3% volume increase will promptly identify whether the volume change has caused a reduction of boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the in-containment refueling water storage tank (IRWST), because the water contained in the IRWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 6).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the pressurizer pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, reduced accumulator capacity might be available for injection following a DBA that required operation of the accumulators. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.5.1.5 (continued)

This SR allows power to be supplied to the motor operated isolation valves when pressurizer pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startup or shutdowns.

Should closure of a valve occur, the safeguard actuation signal provided to the valve would open a closed valve, if required.

SR 3.5.1.6

This SR requires performance of a system performance test of each accumulator to verify flow capabilities. The system performance test demonstrates that the accumulator injection line resistance assumed in accident analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

REFERENCES

1. IEEE Standard 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations."
2. Section 6.3 "Passive Core Cooling System."
3. Section 15.6 "Decrease in Reactor Coolant Inventory."
4. AP1000 PRA Success Criteria.
5. 10 CFR 50.46.
6. NUREG-1366, February 1990.
7. Regulatory Guide 1.177, 8/98, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.2 Core Makeup Tanks (CMTs) - Operating

BASES

BACKGROUND

Two redundant CMTs provide sufficient borated water to assure Reactor Coolant System (RCS) reactivity and inventory control for all design basis accidents (DBAs), including both loss of coolant accident (LOCA) events and non-LOCA events (Ref. 1).

The CMTs are cylindrical tanks with hemispherical upper and lower heads. They are made of carbon steel and clad on the internal surfaces with stainless steel. They are located in containment at an elevation slightly above the reactor coolant loops. Each tank is full of borated water at > 3400 ppm. During normal operation, the CMTs are maintained at RCS pressure through a normally open pressure balance line from the cold leg.

The outlet line from each CMT is connected to one of two direct vessel injection lines, which provides an injection path for the water supplied by the CMT. The outlet line from each CMT is isolated by parallel, normally closed, fail open valves. Upon receipt of a safeguards actuation signal, these four valves open to align the CMTs to the RCS.

The CMTs will inject to the RCS as inventory is lost and steam or reactor coolant is supplied to the CMT to displace the water that is injected. Steam or reactor coolant is provided to the CMT through the cold leg balance line, depending upon the specific event that has occurred. The inlet line from the cold leg is sized for LOCA events, where the cold legs become voided and higher CMT injection flows are required.

The injection line from each CMT contains a flow tuning orifice that is used to provide a mechanism for the field adjustment of the injection line resistance. The orifice is used to establish the required flow rates for the associated plant conditions assumed in the CMT design. The CMT flow is based on providing injection for a minimum of 20 minutes after CMT actuation.

The CMT size and injection capability are selected to provide adequate RCS boration and safety injection for the limiting DBA. One CMT is adequate for this function during

(continued)

BASES

BACKGROUND
(continued)

a small break LOCA where one CMT completely spills via the pipe break (Ref. 2). The Probabilistic Risk Assessment (PRA) (Ref. 3) shows that none of the CMTs are required for small LOCAs, assuming that at least one accumulator is available.

APPLICABLE
SAFETY ANALYSES

The CMTs are assumed to be OPERABLE to provide emergency boration and core makeup when the Chemical and Volume Control System (CVS) is inoperable, and to mitigate the consequences of any DBA which requires the safety injection of borated water (Ref. 2).

Following a non-LOCA event such as a steamline break, the RCS experiences a decrease in temperature and pressure due to an increase in energy removal by the secondary system. The cooldown results in a reduction of the core SHUTDOWN MARGIN due to the negative moderator temperature coefficient, with a potential for return to power. The actuation of the CMTs following this event provides injection of borated water to mitigate the reactivity transient and ensure the core remains shut down.

In the case of a steam generator tube rupture (SGTR), CMT injection provides borated water to compensate for RCS LEAKAGE.

In the case of an RCS leak of 10 gallons per minute, the CMTs can delay depressurization for at least 10 hours, providing makeup to the RCS and remain able to provide the borated water to compensate for RCS shrinkage and to assure the RCS boration for a safe shutdown.

In the case of a LOCA, the CMTs provide a relatively large makeup flow rate for approximately 20 minutes, in conjunction with the accumulators to provide the initial core cooling.

CMTs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO establishes the minimum conditions necessary to ensure that sufficient CMT flow will be available, assuming the minimum requirements of all other PXS components are met, to meet the initial conditions assumed in the safety

(continued)

BASES

LCO
(continued)

analyses. The volume of each CMT represents 100% of the total injected flow assumed in LOCA analysis. If the injection line from a single CMT to the vessel breaks, no single active failure on the other CMT will prevent the injection of borated water into the vessel. Thus the assumptions of the LOCA analysis will be satisfied.

For non-LOCA analysis, two CMTs are assumed. Note that for non-LOCA analysis, the accident cannot disable a CMT.

APPLICABILITY

In MODES 1, 2, 3, and 4 when the RCS is not being cooled by the Normal Residual Heat Removal System (RNS) the CMTs are required to be OPERABLE to provide borated water for RCS inventory makeup and reactivity control following a design basis event and subsequent cooldown.

The CMT requirements in MODE 5 with the RCS pressure boundary intact are specified in LCO 3.5.3, "Core Makeup Tanks (CMTs) - Shutdown, RCS Intact."

The CMTs are not required to be OPERABLE while in MODE 5 with the RCS pressure boundary open or in MODE 6 because the RCS is depressurized and borated water can be supplied from the In-containment Refueling Water Storage Tank (IRWST), if needed.

In the unlikely event of a total loss of AC power sources, coupled with an inoperable Passive Residual Heat Removal Heat Exchanger (PRHR HX) (beyond DBA), the CMTs may be used in a feed and bleed sequence to remove heat from the RCS.

ACTIONS

A.1

With one outlet isolation valve inoperable on one CMT, action must be taken to restore the valve. In this Condition, the CMT is capable of performing its safety function, provided a single failure of the remaining parallel isolation valve does not occur. A Completion Time of 72 hours is acceptable for two train ECCS systems which are capable of performing their safety function without a single failure.

(continued)

BASES

ACTIONS
(continued)

B.1

If the water temperature or boron concentration of one CMT is not within limits, it must be returned to within limits within 72 hours. The deviations in these parameters are expected to be slight, considering the frequent surveillances and control room monitors. With the temperature above the limit, the full core cooling capability assumed in the safety analysis may not be available. With the boron concentration not within limits, the ability to maintain subcriticality following a DBA may be degraded. However, because only one of two CMTs is inoperable, and the deviations of these parameters are expected to be slight, it is probable that more than a required amount of boron and cooling capability will be available to meet the conditions assumed in the safety analysis.

Since the CMTs are redundant, safety class components, the 72 hour Completion Time is consistent with the times normally allowed for this type of component.

C.1

With two CMTs inoperable due to water temperature or boron concentration, at least one CMT must be restored to within limits in 8 hours. The deviations in these parameters are expected to be slight, considering the frequent surveillances and control room monitors. A Completion Time of 8 hours is considered acceptable since the CMTs are expected to be capable of performing their safety function with slight deviations in these parameters.

D.1

Excessive amounts of noncondensable gases in a CMT inlet line may interfere with the natural circulation flow (hot water from the RCS through the balance line into the CMT and cold water from the CMT through the direct vessel injection line into the vessel) assumed in the safety analyses for some transients. For CMT injection following a LOCA (steam will enter the CMT through the balance line, displacing the CMT water), gases in the CMT inlet line are not detrimental to the CMT function. The presence of some noncondensable gases does not mean that the CMT natural circulation capability is immediately inoperable, but that gases are collecting and should be vented. The venting of these gases requires containment entry to manually operate the vent

(continued)

BASES

ACTIONS

D.1 (continued)

valves. A Completion Time of 24 hours is permitted for venting noncondensable gases and is acceptable, since, for the transients, the natural circulation capability of one CMT is adequate to ensure mitigation assuming less conservative analysis assumptions regarding stuck rods and core characteristics.

E.1

With one CMT inoperable for reasons other than Condition A, B, C, D, operation of the CMT may not be available. Action must be taken to restore the inoperable CMT to OPERABLE status within 8 hours. The remaining CMT is sufficient for DBAs except for LOCA in the OPERABLE CMTs DVI line. The 8 hour Completion Time is based on the availability of injection from the accumulators and IRWST lines to provide SI injection. The effectiveness of accumulator and IRWST injection is demonstrated in analysis performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a small LOCA these other injection supplies provide adequate core cooling.

This analysis provides a high confidence that with the unavailability of one CMT the core can be cooled following design bases accidents.

F.1 and F.2

If the Required Action or associated Completion Time of Condition A, B, C, D, or E are not met or the LCO is not met for reasons other than Conditions A through E, 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.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.1 and SR 3.5.2.2

Verification every 24 hours and 7 days that the temperature and the volume, respectively, of the borated water in each CMT is within limits ensures that when a CMT is needed to inject water into the RCS, the injected water temperature and volume will be within the limits assumed in the accident analysis. The 24 hour Frequency is adequate, based on the fact that no mechanism exists to rapidly change the temperature of a large tank of water such as a CMT. These parameters are normally monitored in the control room by indication and alarms. Also, there are provisions for monitoring the temperature of the inlet and outlet lines to detect in-leakage which may affect the CMT water temperature.

SR 3.5.2.3

Each CMT inlet isolation valve must be verified to be fully open each 12 hours. Frequent verification is considered to be important, since a CMT can not perform its safety function, if the valve is closed. Control room instrumentation is normally available for this verification.

SR 3.5.2.4

Verification that excessive amounts of noncondensable gases are not present in the inlet line is required every 24 hours. The inlet line of each CMT has a vertical section of pipe which serves as a high point collection point for noncondensable gases. Control room indication of the water level in the high point collection point is available to verify that noncondensable gases have collected to the extent that the water level is depressed below the allowable level. The 24 hour Frequency is based on the expected low rate of gas accumulation and the availability of control room indication.

SR 3.5.2.5

Verification every 7 days that the boron concentration in each CMT is within the required limits ensures that the reactivity control from each CMT, assumed in the safety

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.5 (continued)

analysis, will be available as required. The 7 day Frequency is adequate to promptly identify changes which could occur from mechanisms such as in-leakage.

SR 3.5.2.6

Verification that the redundant outlet isolation valves are OPERABLE by stroking the valves open ensures that each CMT will function as designed when these valves are actuated. Prior to opening the outlet isolation valves, the inlet isolation valve should be closed temporarily. Closing the inlet isolation valve ensures that the CMT contents will not be diluted or heated by flow from the RCS. Upon completion of the test, the inlet isolation valves must be opened. The Surveillance Frequency references the inservice testing requirements.

SR 3.5.2.7

This SR requires performance of a system performance test of each CMT to verify flow capabilities. The system performance test demonstrates that the CMT injection line resistance assumed in DBA analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

REFERENCES

1. Section 6.3, "Passive Core Cooling System."
 2. Chapter 15, "Accident Analysis."
 3. AP1000 PRA Success Criteria.
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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.3 Core Makeup Tanks (CMTs) – Shutdown, RCS Intact

BASES

BACKGROUND A description of the CMTs is provided in the Bases for LCO 3.5.2, "Core Makeup Tanks – Operating."

APPLICABLE SAFETY ANALYSES When the plant is shutdown with the Reactor Coolant System (RCS) pressure boundary intact, the CMT and Passive Residual Heat Removal (PRHR) are the preferred methods for mitigation of postulated events such as loss of normal decay heat removal capability (either loss of Startup Feedwater or loss of normal residual heat removal system). The CMT and PRHR are preferred because the RCS pressure boundary can remain intact, thus preserving one of the barriers to fission product release. For these events, the PRHR provides the safety related heat removal path. And the CMT maintains RCS inventory control. These events can also be mitigated by In-containment Refueling Water Storage Tank (IRWST) injection; however, the RCS must be depressurized (vented) in order to facilitate IRWST injection.

Since no loss of coolant accidents (LOCAs) are postulated during MODES 5 and 6, the possibility of a break in the direct vessel injection line is not considered. As a result, only one CMT is required to be available to provide core cooling in response to postulated events. The two parallel CMT outlet isolation valves ensure that injection from one CMT occurs in the event of a single active failure.

CMTs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO This LCO establishes the minimum conditions necessary to ensure that one CMT will be available for RCS inventory control in the event of the loss of normal decay heat removal capability. The two CMT outlet isolation valves must be OPERABLE to ensure that at least one valve will operate, assuming that the other valve is disabled by a single active failure.

(continued)

BASES (continued)

APPLICABILITY In MODE 4 without steam generator heat removal and in MODE 5 with the RCS pressure boundary intact, one CMT is required to provide borated water to the RCS in the event the nonsafety related chemical and volume control system makeup pumps are not available to provide RCS inventory control.

The CMT requirements in MODES 1, 2, 3, and 4 are specified in LCO 3.5.2, "Core Makeup Tanks (CMTs) – Operating."

The CMTs are not required to be OPERABLE while in MODE 5 with the RCS open or in MODE 6 because the RCS is depressurized and borated water can be supplied from the IRWST, if needed.

ACTIONS

A.1

With one outlet isolation valve inoperable action must be taken to restore the valve. In this Condition the CMT is capable of performing its safety function, provided a single failure of the remaining parallel isolation valve does not occur. A Completion Time of 72 hours is consistent with times normally applied to an ECCS system which is capable of performing its safety function without a single failure.

B.1

If the water temperature or boron concentration in the CMT is not within limits, it must be returned to within limits within 72 hours. With the temperature above the limit the makeup capability assumed in the safety analysis may not be available. With the boron concentration not within limits, the ability to maintain subcriticality may be degraded.

Because the mechanisms for significantly altering these parameters in the CMT are limited, it is probable that more than the required amount of boron and cooling capacity will be available to meet the conditions assumed in the safety analysis. Therefore, the 72 hour Completion Time is acceptable.

(continued)

BASES

ACTIONS
(continued)

C.1

With the required CMT inoperable for reasons other than Condition A or B operation of the CMT may not be available. Action must be taken to restore the inoperable CMT to OPERABLE status within 8 hours. LOCAs are not postulated during the MODEs when this LCO is applicable. The only safety function is to provide leakage makeup in case normal RCS makeup is unavailable. The 8 hour Completion Time is based on the availability of injection from the IRWST to provide RCS makeup. The ability of the IRWST to provide RCS injection is demonstrated by analysis performed to show that IRWST injection together with ADS venting provides adequate core cooling. Such analysis was performed for the loss of RNS cooling during midloop operations. The analysis was performed in support of the AP1000 PRA (Ref. 2).

D.1

If the Required Action or associated Completion Time of Conditions A, B, or C are not met or the LCO is not met for reasons other than Conditions A through C, action must be initiated, immediately, to place the plant in a MODE where this LCO does not apply. Action must be initiated, immediately, to place the plant in MODE 5 with RCS pressure boundary open and $\geq 20\%$ pressurizer level. In this condition, core cooling and RCS makeup are provided by IRWST injection and sump recirculation. Opening of the ADS valves ensures that IRWST injection can occur.

SURVEILLANCE
REQUIREMENTS

SR 3.5.3.1

The LCO 3.5.2 Surveillance Requirements (SR 3.5.2.1 through 3.5.2.7) are applicable to the CMT required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.5.2 for a discussion of each SR.

REFERENCES

1. Section 6.3, "Passive Core Cooling System."
 2. AP1000 PRA Success Criteria.
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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.4 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating

BASES

BACKGROUND

The normal heat removal mechanism is the steam generators, which are supplied by the startup feedwater system. However, this path utilizes non-safety related components and systems, so its failure must be considered. In the event the steam generators are not available to remove decay heat for any reason, including loss of startup feedwater, the heat removal path is the PRHR HX (Ref. 1).

The principle component of the PRHR HX is a 100% capacity heat exchanger mounted in the In-containment Refueling Water Storage Tank (IRWST). The heat exchanger is connected to the Reactor Coolant System (RCS) by a inlet line from one RCS hot leg, and an outlet line to the associated steam generator cold leg channel head. The inlet line to the passive heat exchanger contains a normally open, motor operated isolation valve. The outlet line is isolated by two parallel, normally closed air operated valves, which fail open on loss of air pressure or control signal. There is a vertical collection point at the top of the common inlet piping high point which serves as a gas collector. It is provided with level detectors that indicate when noncondensable gases have collected in this area. There are provisions to manually vent these gases to the IRWST.

In order to preserve the IRWST water for long term PRHR HX operation, a gutter is provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation any water collected by the gutter is directed to the normal containment sump. During PRHR HX operation, redundant series air operated valves are actuated to block the draining of condensate to the normal sump and to force the condensate into the IRWST. These valves fail closed on loss of air pressure or control signal.

The PRHR HX size and heat removal capability is selected to provide adequate core cooling for the limiting non-LOCA heatup Design Basis Accidents (DBAs) (Ref. 2). The Probability Risk Assessment (PRA) (Ref. 3) shows that PRHR HX is not required assuming that passive feed and bleed is available. Passive feed and bleed uses the Automatic

Depressurization System (ADS) for bleed and the
CMTs/accumulators/IRWST for feed.

APPLICABLE
SAFETY ANALYSES

In the event of a non-LOCA DBA during normal operation, the PRHR HX is automatically actuated to provide decay heat removal path in the event the normal path through the steam generators is not available (Ref. 2).

The non-LOCA events which establish the PRHR HX parameters are those involving a decrease in heat removal by the

(continued)

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

secondary system, such as loss of main feedwater or other failure in the feedwater system. Since the PRHR HX is passive, it will mitigate the consequences of these events with a complete loss of all AC power sources. The PRHR HX actuates when the CMTs are actuated during LOCA events.

The PRHR HX satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the PRHR HX be OPERABLE so that it can respond appropriately to the DBAs which may require its operation. Since this is a passive component, it does not require the actuation of active components such as pumps for its operability and will be OPERABLE if the inlet valves are in their normally open position, and the normally closed, fail open outlet valves open on receipt of an actuation signal.

In addition to the appropriate valve configuration, operability may be impaired by flow blockage caused by noncondensable gases collecting in the system. Thus the absence of noncondensable gases in the high point is necessary for system operability.

The note requires a reactor coolant pump (RCP) to be operating in the loop with the PRHR HX, Loop 1, if any RCPs are operating. If RCPs are only operating in Loop 2 and no RCPs are operating in Loop 1, there is a possibility there may be reverse flow in the PRHR HX.

APPLICABILITY

The PRHR HX must be OPERABLE in MODES 1, 2, 3, and 4 with the RCS not cooled by the Normal Residual Heat Removal System (RNS) if a plant cooldown is required and the normal cooldown path is not available. Under these conditions, the PRHR HX may be actuated to provide core cooling and to mitigate the consequences of a DBA.

The PRHR HX requirements in MODE 4 with RCS cooling provided by the RNS and in MODE 5 with the RCS pressure boundary intact are specified in LCO 3.5.5, "Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Shutdown, RCS Intact."

The PRHR HX is not capable of natural circulation cooling of the RCS in MODE 5 with the RCS pressure boundary open or in MODE 6.

(continued)

BASES (continued)

ACTIONS

A.1

The outlet line from the PRHR HX is controlled by a pair of normally closed, fail open, air operated valves, arranged in parallel. Thus they are redundant and, if either valve is OPERABLE, the system can function at 100% capacity, assuming other operability conditions are met.

If one valve is inoperable, a Completion Time of 72 hours has been allowed to restore the inoperable valve(s) to OPERABLE status. This Completion Time is consistent with the Completion Times specified for other parallel redundant safety related systems.

B.1

With one air operated IRWST gutter isolation valve inoperable, the remaining isolation valve can function to drain the gutter to the IRWST. Action must be taken to restore the inoperable gutter isolation valve to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable based on the capability of the remaining valve to perform 100% of the required safety function assumed in the safety analyses.

C.1

Excessive amounts of noncondensable gases in the PRHR HX inlet line may interfere with the natural circulation flow of reactor coolant through the PRHR HX. The presence of some noncondensable gases does not mean that the PRHR HX is immediately inoperable, but that gases are collecting and should be vented. The venting of these gases requires containment entry to manually operate the appropriate vent valves. A Completion Time of 24 hours is acceptable considering that passive feed and bleed cooling is available to remove heat from the RCS.

D.1 and D.2

If any of the above Required Actions have not been accomplished in the required Completion Time or the LCO is not met for reasons other than Conditions A, B, or C, the plant must be brought to a MODE in which the LCO does not

(continued)

BASES

ACTIONS

D.1 (continued)

apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4, with the RCS cooled by the RNS, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

With the LCO not met for reasons other than Condition A, B, or C, the PRHR HX must be restored within 8 hours. The 8 hour Completion Time is based on the availability of passive feed and bleed cooling to provide RCS heat removal. The effectiveness of feed and bleed cooling has been demonstrated in analysis and evaluations performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a range of events including loss of main feedwater, SGTR, and small LOCA (as small as 1/2") that feed and bleed cooling provides adequate core cooling.

These analyses and evaluations provide a high confidence that with the unavailability of the PRHR HX the core can be cooled following design bases accidents.

F.1 and F.2

If the PRHR HX is not restored in accordance with Action E.1 within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 5 within 36 hours.

Action F.1 is modified by a Note which requires that prior to initiating cooldown of the plant to MODE 3, redundant means of providing SG feedwater be verified as operable. Possible means include main feedwater and startup feedwater pumps. With the PRHR HX and redundant means of feeding the SGs INOPERABLE, the unit is in a seriously degraded condition with no means for conducting a controlled cooldown. In

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. If redundant means of feeding the SGs are not available, the plant should be maintained in the current MODE until redundant means are restored. LCO 3.0.3 and all other Required Actions shall be suspended until the redundant means are restored, because they could force the unit into a less safe condition.

Action F.2 is modified by a Note which requires that prior to stopping SG feedwater, redundant means of cooling the RCS to cold shutdown conditions must be verified as operable. One redundant means of cooling the RCS to cold shutdown includes the normal residual heat removal system (RNS) and its necessary support system (both component cooling system pumps and heat exchangers, and both service water system pumps and fans). Without availability of these redundant cooling means, the unit is in a seriously degraded condition with no means for continuing the controlled cooldown. Until the redundant cooling means are restored, heat removal using SG feedwater should be maintained. LCO 3.0.3 and all other Required Actions shall be suspended until the systems and equipment required for further cooldown are restored, because they could force the unit into a less safe condition.

SURVEILLANCE
REQUIREMENTS

SR 3.5.4.1

Verification, using remote indication, that the common outlet manual isolation valve is fully open ensures that the flow path from the heat exchangers to the RCS is available. Misalignment of this valve could render the heat exchanger inoperable. A 12 hour Frequency is reasonable considering that the valve is manually positioned and has control room position indication and alarm.

SR 3.5.4.2

Verification that the motor operated inlet valve is fully open, as indicated in the main control room, ensures timely discovery if the valve is not fully open. The 12 hour

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.5.4.2 (continued)

Frequency is consistent with the ease of verification, confirmatory open signals, and redundant series valve controls that prevent spurious closure.

SR 3.5.4.3

Verification that excessive amounts of noncondensable gases are not present in the inlet line is required every 24 hours. The inlet line of the PRHR HX has a vertical section of pipe which serves as a high point collection point for noncondensable gases. Control room indication of the water level in this high point collection point is available to verify that noncondensable gases have not collected to the extent that the water level is depressed below the allowable level. The 24 hour Frequency is based on the expected low rate of gas accumulation and the availability of control room indication.

SR 3.5.4.4

Verification is required to confirm that power is removed from the motor operated inlet isolation valve every 31 days. Removal of power from this valve reduces the likelihood that the valve will be inadvertently closed as a result of a fire. The 31 day Frequency is acceptable considering the frequent surveillance of valve position and that the valve has a confirmatory open signal.

SR 3.5.4.5

Verification that both air operated outlet valves and both IRWST gutter isolation valves are OPERABLE ensures that the PRHR HX will actuate on command, with return flow from the gutter to the IRWST, since all other components of the system are normally in the OPERABLE configuration. Since these valves are redundant, if one valve is inoperable, the system can function at 100% capacity. Verification requires the actual operation of each valve through a full cycle to demonstrate operability. The Surveillance Frequency is provided in the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.4.6

This SR requires performance of a system performance test of the PRHR HX to verify system heat transfer capabilities. The system performance test demonstrates that the PRHR HX heat transfer assumed in accident analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

SR 3.5.4.7

This surveillance requires visual inspection of the IRWST gutters to verify that the return flow to the IRWST will not be restricted by debris. A Frequency of 24 months is adequate, since there are no known sources of debris with which the gutters could become restricted.

REFERENCES

1. Section 6.3, "Passive Core Cooling System."
 2. Chapter 15, "Safety Analysis."
 3. AP1000 PRA Success Criteria.
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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.5 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Shutdown,
RCS Intact

BASES

BACKGROUND A description of the PRHR HX is provided in the Bases for LCO 3.5.4, "Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating."

APPLICABLE SAFETY ANALYSES In the event of a loss of normal decay heat removal capability during shutdown with the Reactor Coolant System (RCS) pressure boundary intact, the PRHR HX provides the preferred safety related heat removal path. When required, the PRHR HX is manually actuated and can maintain the RCS < 420°F. Alternatively, the heat removal function can be provided by depressurizing the RCS with the Automatic Depressurization System (ADS) and injection of the In-containment Refueling Water Storage Tank (IRWST) with containment closure capability provided. The PRHR HX is preferred because the RCS pressure boundary remains intact, thus preserving a barrier to fission product release.

The PRHR HX satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO This LCO requires the PRHR HX to be OPERABLE so that it can be placed in service in the event normal decay heat removal capability is lost. Since this a passive component, it does not require the actuation of active components such as pumps for its operability and will be OPERABLE if the inlet valves are in their normally open position, and the normally closed, fail open outlet valves open on receipt of an actuation signal.

In addition to the appropriate valve configuration, operability may be impaired by flow blockage caused by noncondensable gases collecting in the system. Thus the absence of non-condensable gases in the high point is necessary for system operability.

The note requires a reactor coolant pump (RCP) to be operating in loop one if any RCPs are operating. If RCPs

(continued)

BASES

LCO
(continued) are only operating in loop 2 and no RCPs are operating in loop 1, there is a possibility there may be reverse flow in the PRHR HX.

APPLICABILITY The PRHR HX must be OPERABLE in MODE 4 with RCS cooling provided by the RNS and in MODE 5 with the RCS pressure boundary intact to provide decay heat removal in the event the normal residual heat removal system is not available.

The PRHR HX requirements in MODES 1, 2, 3, and 4 with RCS cooling not provided by the RNS are specified in LCO 3.5.4, "Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating."

The PRHR HX is not capable of natural circulation cooling of the RCS in MODE 5 with the RCS pressure boundary open or in MODE 6.

ACTIONS

A.1

The outlet line from the PRHR HX is isolated by a pair of normally closed, fail open, air operated valves, arranged in parallel. They are redundant, and if either valve is OPERABLE the system can function at 100% capacity, assuming other operability conditions are met.

Since these valves are redundant, if one valve is inoperable, a Completion Time of 72 hours has been allowed to restore the inoperable valve to OPERABLE status. This Completion Time is consistent with the Completion Times specified for other parallel redundant safety related systems.

B.1

With one air operated IRWST gutter isolation valve inoperable, the remaining isolation valve can function to drain the gutter to the IRWST. Action must be taken to restore the inoperable gutter isolation valve to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable based on the capability of the remaining valve to perform 100% of the required safety function assumed in the safety analyses.

(continued)

BASES

ACTIONS
(continued)

C.1

At the inlet piping high point there is a vertical chamber which serves as a collection point for noncondensable gases. This collection point is provided with detectors which alarm to indicate when gases have collected in this area. The presence of an alarm does not mean that PRHR HX is immediately inoperable, but that gases are collecting and should be vented. A Completion Time of 24 hours is acceptable, considering that passive feed and bleed cooling is available to remove heat from the RCS.

D.1

With the LCO not met for reasons other than Condition A, B, or C, the PRHR HX must be restored within 8 hours. The 8 hour Completion Time is acceptable based on the availability of passive feed and bleed cooling to provide RCS heat removal. The effectiveness of feed and bleed cooling is discussed in the bases for LCO 3.4.4, Action E.1.

E.1

If any of the above Required Actions have not been accomplished in the required Completion Time, or the LCO is not met for reasons other than Conditions A, B, C, or D, action must be initiated, immediately, to be in MODE 5 with the RCS pressure boundary open and pressurizer level $\geq 20\%$. The time to RCS boiling is maximized in the event of loss of normal decay heat removal capability, by maintaining a visible level in the pressurizer. Additionally, in this MODE the RCS must be opened, such that safety related decay heat removal can be immediately initiated by actuation of the IRWST injection valve(s).

SURVEILLANCE
REQUIREMENTS

SR 3.5.5.1

The LCO 3.5.4 Surveillance Requirements are applicable to the PRHR HX required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.5.4 for a discussion of each SR.

REFERENCES

None.

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.6 In-containment Refueling Water Storage Tank (IRWST) – Operating

BASES

BACKGROUND

The IRWST is a large stainless steel lined tank filled with borated water (Ref. 1). It is located below the operating deck in containment. The tank is designed to meet seismic Category 1 requirements. The floor of the IRWST is elevated above the reactor coolant loop so that borated water can drain by gravity into the Reactor Coolant System (RCS). The IRWST is maintained at ambient containment pressure.

The IRWST has two injection flow paths. The injection paths are connected to the reactor vessel through two direct vessel injection lines which are also used by the accumulators and the core makeup tanks. Each path includes an injection flow path and a containment recirculation flow path. Each injection path includes a normally open motor operated isolation valve and two parallel lines each isolated by one check valve and one squib valve in series.

The IRWST has two containment recirculation flow paths. Each containment recirculation path contains two parallel flow paths, one path is isolated by a normally open motor operated valve in series with a squib valve and one path is isolated by a check valve in series with a squib valve.

During refueling operations, the IRWST is used to flood the refueling cavity. During abnormal events, the IRWST serves as a heat sink for the passive residual heat removal heat exchangers, as a heat sink for the depressurization spargers, and as a source of low head (ambient containment pressure) safety injection during loss of coolant accidents (LOCAs) and loss of decay heat removal in MODE 5 (loops not filled). The IRWST can be cooled by the Normal Residual Heat Removal System (RNS) system.

The IRWST size and injection capability is selected to provide adequate core cooling for the limiting Design Basis Accidents (DBAs) (Ref. 2).

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

During non-LOCA events, the IRWST serves as the initial heat sink for the PRHR Heat Exchanger (PRHR HX) if used during reactor cooldown to MODE 4. If RNS is available, it will be actuated in MODE 4 and used to continue the plant cooldown to MODE 5. If RNS is not available, cooldown can continue on PRHR. Continued PRHR HX operation will result in the water in the IRWST heating up to saturation conditions and boiling. The steam generated in the IRWST enters the containment through the IRWST vents. Most of the steam generated in the IRWST condenses on the inside of the containment vessel and drains back to the IRWST.

For events which involve a loss of primary coolant inventory, such as a large break LOCA, or other events involving automatic depressurization, the IRWST provides low pressure safety injection (Ref. 2). The IRWST drain down time is dependent on several factors, including break size, location, and the return of steam condensate from the passive containment cooling system. During drain down, when the water in the IRWST reaches the Low 5 level, the containment sump will be sufficiently flooded, to initiate containment sump recirculation. This permits continued cooling of the core by recirculation of the spilled water in the containment sumps via the sump recirculation flow paths. In this situation, core cooling can continue indefinitely.

When the plant is in midloop operation, the pressurizer Automatic Depressurization System (ADS) valves are open, and the RNS is used to cool the RCS. The RNS is not a safety related system, so its failure must be considered. In this situation, with the RCS drained and the pressure boundary open, the PRHR HX cannot be used. In such a case, core cooling is provided by gravity injection from the IRWST, venting the RCS through the ADS. Injection from the IRWST provides core cooling until the tank empties and gravity recirculation from the containment starts. With the containment closed, the recirculation can continue indefinitely, with the decay heat generated steam condensing on the containment vessel and draining back into the IRWST.

The IRWST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO The IRWST requirements ensure that an adequate supply of borated water is available to act as a heat sink for PRHR and to supply the required volume of borated water as safety injection for core cooling and reactivity control.

To be considered OPERABLE, the IRWST must meet the water volume, boron concentration, and temperature limits defined in the surveillance requirements. The motor operated injection isolation valves must be open with power removed, and the motor operated sump recirculation isolation valves must be open.

APPLICABILITY In MODES 1, 2, 3, and 4, a safety related function of the IRWST is to provide a heat sink for PRHR. In MODES 1, 2, 3, 4, and 5, a second safety related function is the low head safety injection of borated water following a LOCA for core cooling and reactivity control. Both of these functions must be available to meet the initial assumptions of the safety analyses. These assumptions require the specified boron concentration, the minimum water volume, and the maximum water temperature.

The requirements for the IRWST in MODES 5 and 6 are specified in LCO 3.5.7, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5 and LCO 3.5.8, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6.

ACTIONS

A.1

If a motor operated containment sump isolation valve in one sump recirculation flow path is not fully open, the valve must be fully opened within 72 hours. In this condition, three other sump recirculation flow paths are available and can provide 100% of the required flow assuming a break in the direct vessel injection line associated with the other injection train, but with no single failure of the check valve flow path in the same sump recirculation flow path. The 72 hour Completion Time is consistent with times normally applied to degraded two train ECCS systems which can provide 100% of the required flow without a single failure.

(continued)

BASES

ACTIONS
(continued)

B.1

If the IRWST water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR HX heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The long term cooling analysis performed to justify PRA success criteria (Ref. 3) assumed multiple failures with as many as 3 CMTs/Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8 hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to DBAs with slight deviations in these parameters.

C.1

If the motor operated IRWST isolation valves are not fully open, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valves must be restored to fully open in 1 hour. This Completion Time is acceptable based on risk considerations.

D.1 and D.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Times or the LCO is not met for reasons other than Conditions A, B, or C, 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.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.5.6.1

The IRWST borated water temperature must be verified every 24 hours to ensure that the temperature is within the limit assumed in the accident analysis. This Frequency is sufficient to identify a temperature change that would approach the limit and has been shown to be acceptable through operating experience.

SR 3.5.6.2

Verification every 24 hours that the IRWST borated water volume is above the required minimum level will ensure that a sufficient initial supply is available for safety injection and floodup volume for recirculation and as the heat sink for PRHR. During shutdown with the refueling cavity flooded with water from the IRWST, this Surveillance requires that the combined volume of borated water in the IRWST and refueling cavity meet the specified limit. Since the IRWST volume is normally stable, and is monitored by redundant main control indication and alarm, a 24 hour Frequency is appropriate.

SR 3.5.6.3

Verification every 31 days that the boron concentration of the IRWST is greater than the required limit, ensures that the reactor will remain subcritical following a LOCA. Since the IRWST volume is large and normally stable, the 31 day Frequency is acceptable, considering additional verifications are required within 6 hours after each solution volume increase of 15,000 gal. In addition, the relatively frequent surveillance of the IRWST water volume provides assurance that the IRWST boron concentration is not changed.

SR 3.5.6.4

This surveillance requires verification that each motor operated outlet isolation valve is fully open. This surveillance may be performed with available remote position indication instrumentation. The 12 hour Frequency is acceptable, considering the redundant remote indication and alarms and that power is removed from the valve operator.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.6.5

Verification is required to confirm that power is removed from each motor operated IRWST outlet isolation valve each 31 days. Removal of power from these valves reduces the likelihood that the valves will be inadvertently closed. The 31 day Frequency is acceptable considering frequent surveillance of valve position and that the valve has a confirmatory open signal.

SR 3.5.6.6

Each motor operated containment recirculation isolation valve must be verified to be fully open. This valve is required to be open to improve containment recirculation reliability. The 31 day Frequency is acceptable considering the valve has a confirmatory open signal. This surveillance may be performed with available remote position indication instrumentation.

SR 3.5.6.7

This Surveillance requires verification that each IRWST injection and each containment recirculation squib valve is OPERABLE in accordance with the Inservice Testing Program. The Surveillance Frequency for verifying valve OPERABILITY references the Inservice Testing Program.

The squib valves will be tested in accordance with ASME Section XI which specifies valve testing in accordance with the ASME OM Code. The applicable ASME OM Code squib valve requirements are specified in paragraph 4.6, Inservice Tests for Category D Explosively Actuated Valves. The requirements include actuation of a sample of the installed valves each 2 years and periodic replacement of charges.

SR 3.5.6.8

Visual inspection is required each 24 months to verify that the IRWST screens and the containment recirculation screens are not restricted by debris. A Frequency of 24 months is adequate, since there are no known sources of debris with which the gutters could become restricted.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.6.9

This SR requires performance of a system inspection and performance test of the IRWST injection and recirculation flow paths to verify system flow capabilities. The system inspection and performance test demonstrates that the IRWST injection and recirculation capabilities assumed in accident analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

REFERENCES

1. Section 6.3, "Passive Core Cooling."
 2. Section 15.6, "Decrease in Reactor Coolant Inventory."
 3. AP1000 PRA Success Criteria.
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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.7 In-containment Refueling Water Storage Tank (IRWST) – Shutdown,
MODE 5

BASES

BACKGROUND A description of the IRWST is provided in LCO 3.5.6,
"In-containment Refueling Water Storage Tank – Operating."

APPLICABLE SAFETY ANALYSES For postulated shutdown events in MODE 5 with the Reactor Coolant System (RCS) pressure boundary intact, the primary protection is Passive Residual Heat Removal (PRHR), where the IRWST serves as the initial heat sink for the PRHR heat exchanger (PRHR HX). For events in MODE 5 with the RCS pressure boundary open, PRHR is not available and RCS heat removal is provided by IRWST injection and containment sump recirculation.

IRWST injection could be required to mitigate some events by providing RCS inventory makeup.

No loss of coolant accidents (LOCAs) are postulated during plant operation in MODE 5; therefore, the rupture of the direct vessel injection line (DVI) is not assumed. Since the DVI rupture is not assumed, only one train of IRWST injection and recirculation flow paths is required to mitigate postulated events, assuming a single failure.

The IRWST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii):

LCO The IRWST requirements ensure that an adequate supply of borated water is available to act as a heat sink for PRHR and to supply the required volume of borated water as safety injection for core cooling and reactivity control.

To be considered OPERABLE, the IRWST must meet the water volume, boron concentration, and temperature limits defined in the Surveillance Requirements, and one path of injection and recirculation must be OPERABLE (the motor operated injection isolation valve must be open with power removed, and the motor operated sump recirculation isolation valves must be open).

(continued)

BASES (continued)

APPLICABILITY In MODE 5 with the RCS pressure boundary intact or with the RCS open with pressurizer level $\geq 20\%$, the IRWST is an RCS injection source of borated water for core cooling and reactivity control. Additionally, in MODE 5 with the RCS pressure boundary intact, the IRWST provides the heat sink for PRHR.

The requirements for the IRWST in MODES 1, 2, 3, and 4 are specified in LCO 3.5.6, In-containment Refueling Water Storage Tank (IRWST) – Operating. The requirements for the IRWST in MODE 6 are specified in LCO 3.5.8, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6.

ACTIONS

A.1

If a motor operated containment sump isolation valve in one sump recirculation flow path is not fully open, the valve must be fully opened within 72 hours. The 72 hour Completion Time is consistent with times normally applied to degraded two train ECCS systems which can provide 100% of the required flow without a single failure.

B.1

If the IRWST water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The long term cooling analysis performed to justify PRA success criteria (Ref. 3) assumed multiple failures with as many as 3 CMTs/Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8 hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to shutdown events with slight deviations in these parameters.

(continued)

BASES

ACTIONS
(continued)

C.1

If the motor operated IRWST isolation valves are not fully open, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valves must be restored to fully open in 1 hour. This Completion Time is acceptable based on risk considerations.

D.1 and D.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Times or the LCO is not met for reasons other than Conditions A, B, or C, the plant must be placed in a condition in which the probability and consequences of an event are minimized to the extent possible. This is done by immediately initiating action to place the plant in MODE 5 with the RCS intact with $\geq 20\%$ pressurizer level. The time to RCS boiling is maximized by maintaining RCS inventory at $\geq 20\%$ pressurizer level and maintaining RCS temperature as low as practical. With the RCS intact, the availability of the PRHR HX is maintained. Additionally, action to suspend positive reactivity additions is required to ensure that the shutdown margin is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

SURVEILLANCE
REQUIREMENTS

SR 3.5.7.1

The LCO 3.5.6 Surveillance Requirements and Frequencies (SR 3.5.6.1 through 3.5.6.7) are applicable to the IRWST and the flow paths required to be OPERABLE. Refer to the corresponding Bases for LCO 3.5.6 for a discussion of each SR.

REFERENCES

None.

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.8 In-containment Refueling Water Storage Tank (IRWST) – Shutdown,
MODE 6

BASES

BACKGROUND A description of the IRWST is provided in LCO 3.5.6,
"In-containment Refueling Water Storage Tank (IRWST) –
Operating."

APPLICABLE For MODE 6, heat removal is provided by IRWST injection and
SAFETY ANALYSES containment sump recirculation.

IRWST injection could be required to mitigate some events by
providing RCS inventory makeup.

One line with redundant, parallel valves is required to
accommodate a single failure (to open) of an isolation
valve.

The IRWST satisfies Criteria 2 and 3 of 10 CFR
50.36(c)(2)(ii).

LCO The IRWST requirements ensure that an adequate supply of
borated water is available to supply the required volume of
borated water as safety injection for core cooling and
reactivity control.

To be considered OPERABLE, the IRWST in combination with the
refueling cavity must meet the water volume, boron
concentration, and temperature limits defined in the
Surveillance Requirements, and one path of injection and
recirculation must be OPERABLE. The motor operated
injection isolation valve must be open and power removed,
and the motor operated sump recirculation isolation valves
must be closed and OPERABLE. Any cavity leakage should be
estimated and made up with borated water such that the
volume in the IRWST plus the refueling cavity will meet the
IRWST volume requirement.

(continued)

BASES (continued)

APPLICABILITY In MODE 6, the IRWST is an RCS injection source of borated water for core cooling and reactivity control.

The requirements for the IRWST in MODES 1, 2, 3, and 4 are specified in LCO 3.5.6, In-containment Refueling Water Storage Tank (IRWST) - Operating. The requirements for the IRWST in MODE 5 are specified in LCO 3.5.7, In-containment Refueling Water Storage Tank (IRWST) - Shutdown, MODE 5.

ACTIONS

A.1

With one motor operated containment sump isolation valve not fully open, the valve must be fully opened within 72 hours. The 72 hour Completion Time is consistent with times normally applied to degraded two train ECCS systems which can provide 100% of the required flow without a single failure.

B.1

If the IRWST and refueling cavity water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR HX heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The long term cooling analysis performed to justify PRA success criteria (Ref. 3) assumed multiple failures with as many as 3 CMTs/Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8 hour Completion Time is acceptable considering that the IRWST will be fully capable of performing its assumed safety function in response to shutdown events with slight deviations in these parameters.

C.1

If the motor operated IRWST isolation valves are not fully open, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valves must be restored to fully open in 1 hour. This Completion Time is acceptable based on risk considerations.

(continued)

BASES

ACTIONS
(continued)

D.1 and D.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Times or the LCO is not met for reasons other than Conditions A, B, C, or D, the plant must be placed in a Condition in which the probability and consequences of an event are minimized to the extent possible. In MODE 6, action must be immediately initiated to be in MODE 6 with the cavity water level ≥ 23 feet above the top of the reactor vessel flange.

The time to RCS boiling is maximized by maximizing the RCS inventory and maintaining RCS temperature as low as practical. With the RCS intact, another means of removing decay heat is available (the PRHR HX). Additionally, action to suspend positive reactivity additions is required to ensure that the shutdown margin is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS. These Actions place the plant in a condition which maximizes the time to IRWST injection, thus providing time for repairs or application of alternative cooling capabilities.

SURVEILLANCE
REQUIREMENTS

SR 3.5.8.1

The IRWST and refueling cavity borated water temperature must be verified every 24 hours to ensure that the temperature is within the limit assumed in accident analysis. This Frequency is sufficient to identify a temperature change that would approach the limit and has been shown to be acceptable through operating experience.

SR 3.5.8.2

Verification every 24 hours that the IRWST and refueling cavity borated water volume is above the required minimum level will ensure that a sufficient initial supply is available for safety injection and floodup volume for recirculation and as the heat sink for PRHR. During shutdown with the refueling cavity flooded with water from the IRWST, this Surveillance requires that the combined

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.5.8.2 (continued)

volume of borated water in the IRWST and refueling cavity meet the specified limit. Since the IRWST volume is normally stable, and is monitored by redundant main control indication and alarm, a 24 hour Frequency is appropriate.

SR 3.5.8.3

Verification every 31 days that the boron concentration of the IRWST and refueling cavity is greater than the required limit ensures that the reactor will remain subcritical following shutdown events. Since the IRWST volume is large and normally stable, the 31 day Frequency is acceptable, considering additional verifications are required within 6 hours after each solution volume increase of 15,000 gal.

SR 3.5.8.4

LCO 3.5.6 Surveillance Requirements and Frequencies
SR 3.5.6.4 through 3.5.6.8 are applicable to the IRWST and the flow paths required to be OPERABLE. Refer to the corresponding Bases for LCO 3.5.6 for a discussion of each SR.

REFERENCES

None.
