Entergy

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Technical Specification 5.5.12.d

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Report of Changes to Technical Specifications Bases

Dear Sir or Madam:

This report is submitted in accordance with Palisades Technical Specification 5.5.12.d, which requires that changes to the Technical Specifications Bases, implemented without prior Nuclear Regulatory Commission (NRC) approval, be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Enclosure 1 provides a listing of all bases changes since issuance of the previous report, dated August 17, 2006, and identifies the affected sections and nature of the changes. Enclosure 2 provides page change instructions and a copy of the current Technical Specifications Bases List of Effective Pages, Title Page, Table of Contents, and the revised Technical Specification Bases sections listed in Enclosure 1.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

Christopher J. Schwarz
Site Vice President
Palisades Nuclear Plant

Enclosures (2)

CC Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC

ENCLOSURE 1
TECHNICAL SPECIFICATION BASES CHANGE CHRONOLOGY

DATE	AFFECTED SECTION(S)	CHANGE(S)
09/21/06	B 3.4.4 B 3.4.5 B 3.4.6 B 3.4.7 B 3.4.13 B 3.4.17 (added)	Make Bases consistent with NRC-approved Amendment 223, Steam Generator Program changes based on [Technical Specification Task Force] TSTF-449. The Bases were changed to eliminate the reference to the Steam Generator Tube Surveillance Program as the method of establishing steam generator operability. The Bases were also changed to indicate that satisfying the primary to secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met.
02/14/07	B 3.3.3	Removed an editorial error carried over from Standard Technical Specification (STS) implementation, Amendment 189. The STS implementation inadvertently included an extraneous heading that was out of place. The change removed the text that did not need to be included.
05/15/07	B 3.1.7	Updated Special Test Exception reference ANSI/ANS-19.6.1 from 1997 edition to 2005 edition to support a revision to the startup physics testing procedure.
07/18/07	B 3.1.4	Clarified that a control rod is not considered trippable if surveillance requirement 3.1.4.3 is not performed per the frequency. This change clarified the conditions for entry into TS 3.1.4, Condition D.
07/31/07	B 3.4.6 B 3.4.7 B 3.4.8 B 3.9.4 B 3.9.5 B 3.7.6	Clarified separate shutdown cooling heat exchangers are required for each operable shutdown cooling train. Clarified condensate inventory requirement. This change better describes the purpose of the condensate inventory. The inventory requirement is intended to meet decay heat removal requirements for eight hours and not intended to be sufficient to hold the plant in Mode 3 for four hours and then support a cooldown at 75°F per hour to shutdown cooling conditions.

ENCLOSURE 2

REVISED TECHNICAL SPECIFICATIONS BASES
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B 3.4.5, B 3.4.6, B 3.4.7, B 3.4.8, B 3.4.13,
B 3.4.17, B 3.7.6, B 3.9.4, B 3.9.5

102 Pages Follow

TECHNICAL SPECIFICATIONS BASES CHANGES: September 2007
RENEWED FACILITY OPERATING LICENSE DPR-20
DOCKET NO. 50-255
Page Change Instructions

Revise your copy of the Palisades Technical Specifications Bases with the attached revised pages. The revised pages are identified by amendment number or revision date at the bottom of the pages and contain vertical lines in the margin indicating the areas of change.

REMOVE	INSERT
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Section B 3.4.8	Section B 3.4.8
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--	Section B 3.4.17 (Insert after Section B 3.4.16)
Section B 3.7.6	Section B 3.7.6
Section B 3.9.4	Section B 3.9.4
Section B 3.9.5	Section B 3.9.5

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PALISADES PLANT

FACILITY OPERATING LICENSE DPR-20

APPENDIX A

TECHNICAL SPECIFICATIONS

BASES

As Amended Through Amendment No. 223

Revised 09/21/2006

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Alignment

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating rods is an initial assumption in all safety analyses that assume full-length control rod insertion upon reactor trip. Maximum control rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The Palisades Nuclear Plant design criteria contain the applicable criteria for these reactivity and power distribution design requirements (Ref. 1).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod misalignment may cause increased power peaking, due to the asymmetric reactivity distribution, and a reduction in the total available control rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all control rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Control rods are moved by their Control Rod Drive Mechanisms (CRDMs). Each CRDM moves its rod at a fixed rate of approximately 46 inches per minute. Although the ability to move a full-length control rod by its drive mechanism is not an initial assumption used in the safety analyses, it is required to support OPERABILITY. As such, the inability to move a full-length control rod results in that full-length control rod being inoperable.

The control rods are arranged into groups that are radially symmetric. Therefore, movement of the control rod groups does not introduce radial asymmetries in the core power distribution. The shutdown and regulating rods provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods also provide reactivity (power level) control during normal operation and transients.

BASES

BACKGROUND (continued)

The axial position of shutdown and regulating rods is indicated by two separate and independent systems, which are 1) synchro based position indication system, and 2) the reed switch based position indication system.

The synchro based position indication system measures the phase angle of a synchro geared to the CRDM rack. Full control rod travel corresponds to less than 1 turn of the synchro. Each control rod has its own synchro. The Primary Information Processor (PIP) node scans and converts synchro outputs into inches of control rod withdrawal. The resolution of this system is approximately 0.5 inches. Each synchro also has cam operated limit switches that provide input to the matrix indication lights of control rod status indication for various key positions.

The reed switch based position indication system is referred to as the Secondary Position Indication (SPI) system. This system provides a highly accurate indication of actual control rod position, but at a lower precision than the synchros. The reed switches are wired so that the voltage read across the reed switch stack is proportional to rod position. The reed switches are spaced along a tube with a center-to-center spacing distance of 1.5 inches. The resolution of the SPI reed switch stacks is 1.5 inches. The reed switches also provide input to the matrix indication lights that provide control rod status indication for various key positions. To increase the reliability of the system, there are redundant reed switches that prevent false indication in the event an individual reed switch fails.

A control rod position deviation alarm is provided to alert the operator when any two control rods in the same group are more than 8 inches apart. This helps to ensure any control rod misalignments are minimized. The alarm can be generated by either the SPI system or PIP node since the SPI system, in conjunction with the host computer, is redundant to the PIP node in the task of control rod measurements, control rod monitoring, and limit processing.

BASES

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Refs. 3 and 4). The accident analysis defines control rod misoperation as any event, with the exception of sequential group withdrawals, which could result from a single malfunction in the reactivity control systems. For example, control rod misalignment may be caused by a malfunction of the Rod Control System, or by operator error. A stuck rod may be caused by mechanical jamming. Inadvertent withdrawal of a single control rod may be caused by an electrical or mechanical failure in the Rod Control System. A dropped control rod could be caused by an electrical or mechanical failure in the CRDM.

The acceptance criteria for addressing control rod inoperability/misalignment are that:

a. There shall be no violations of:

   1. Specified Acceptable Fuel Design Limits (SAFDL), or

   2. Primary Coolant System (PCS) pressure boundary integrity; and

b. The core must remain subcritical after accident transients.

Three types of misoperations are discussed in the safety analysis (Ref. 4). During movement of a group, one control rod may stop moving while the other control rods in the group continue. This condition may cause excessive power peaking. The second type of misoperations occurs if one control rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining control rods to meet the SDM requirement with the maximum worth rod stuck fully withdrawn. If a control rod is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck rods into account. The third type of misoperations occurs when one rod drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power, due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local Linear Heat Rates (LHRs).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The most limiting static misalignment occurs when Bank 4 is fully inserted with one rod fully withdrawn ([Bank 4 is 99 inches out of alignment with the rated Power Dependent Insertion Limit [PDIL].) This event was bounded by the dropped full-length control rod event (Ref. 4).

Since the control rod drop incidents result in the most rapid approach to SAFDLs caused by a control rod misoperation, the accident analysis analyzed a single full-length control rod drop.

The above control rod misoperations may or may not result in an automatic reactor trip. In the case of the full-length rod drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters.

The results of the control rod misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or PCS pressure occur.

Control rod alignment satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).

LCO

The limits on shutdown, regulating, and part-length rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the full-length control rods will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the control rod banks maintain the correct alignment and that each full-length control rod is capable of being moved by its CRDM. The OPERABILITY requirement for the part-length rods is that they are fully withdrawn.

The requirement is to maintain the control rod alignment to within 8 inches between any control rod and all other rods in its group. To help ensure this requirement is met, the control rod position deviation alarm generated by either the PIP node or the SPI system, must be OPERABLE and provide an alarm when any control rod becomes misaligned > 8 inches from any other rod in its group. The safety analysis assumes a total misalignment from fully withdrawn to fully inserted. This case bounds the safety analysis for a single rod in any intermediate position.

BASES

LCO (continued)

The primary rod position indication system is considered OPERABLE, for purposes of this specification, if the digital position readout or the PPC display provides valid rod position indication, or if the cam operated red matrix light (regulating and part-length rods only) gives positive (ON) indication of rod position. The secondary rod position indication system is considered OPERABLE, for purposes of this specification, if the magnetically operated reed switches are providing valid indication of rod position either via the plant process computer or by taking direct readings of the output from the magnetic reed switches or if the reed switch operated red matrix light (shutdown rods only) gives positive (ON) indication of rod position.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDM, any of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on control rod OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (e.g., trippability) and alignment of control rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and regulating rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the PCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

LCOs 3.1.4, 3.1.5, and 3.1.6, and their ACTIONS were written to support each other. The combined intent is to assure the following:

1. There is adequate SDM available in withdrawn control rods to assure the reactor is shutdown by, and remains shutdown following, a reactor trip,

2. The control rod positioning does not cause unacceptable axial or radial flux peaking, and

3. The programmed rod withdrawal sequence and group overlap result in reactivity insertion rates within the assumptions of the Inadvertent Control Rod Bank Withdrawal Analyses.

BASES

ACTIONS (continued)

The ACTIONS for rods that are mispositioned (misaligned or inserted beyond the limit) were written assuming that an OPERABLE rod discovered to be mispositioned would simply be re-positioned correctly. While the associated Conditions would have to be entered, the rod could be re-positioned (thus exiting the LCO) without taking any other Required Action. A rod that remains mispositioned was assumed to be inoperable. The analyses account for operation with one (and only one) mispositioned rod (a dropped rod being the limiting case). With more than one mispositioned rod, the plant would be outside the bounds of the analyses and must be shutdown.

If a rod is discovered to be misaligned (ie, there is more than 8 inches between it and any other rod in its group, but all remaining rods in that group are within 8 inches of each other) Condition 3.1.4 C allows 2 hours to restore the rod alignment (thus exiting the LCO), perform SR 3.2.2.1 (verification that radial peaking is within limits), or reduce power to ≤ 75% RTP.

If one or more shutdown rods are inserted beyond the insertion limit, Condition 3.1.5 A is entered; the rods are declared inoperable and Condition 3.1.4 D (when one rod is immovable but trippable) or Condition 3.1.4 E (when a movable rod is inserted beyond its insertion limit, or when more than one rod is inoperable for any reason) must be entered.

If the rods can be moved, they should be withdrawn and all Conditions exited.

If one rod cannot be moved (but is still considered trippable), operation may continue in accordance with Condition 3.1.4 D (and 3.1.4 C if it is misaligned).

If more than one rod cannot be moved, Condition 3.1.4 E must also be entered. The plant must be in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

If one or more part length rods are inserted beyond the limit, Condition 3.1.5 A is entered; the rods are declared inoperable and Condition 3.1.4 E is entered (and 3.1.4 C if it is misaligned). Condition 3.1.4 D is not applicable to part-length rods since it only addresses full-length rods.

If the rods can be moved, they should be withdrawn and all Conditions exited.

BASES

ACTIONS (continued)

If any part-length rods are inserted beyond the limit and cannot be moved, the plant must be placed in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

If one or more OPERABLE regulating rods are inserted beyond the limit, Condition 3.1.6 A is entered.

The rods must be restored to within limits (by rod withdrawal or power reduction) within two hours.

If a rod cannot be moved, it must be considered inoperable and Condition 3.1.4 D must be entered (and 3.1.4 C if it is misaligned). Condition 3.1.4 D allows continued operation with one inoperable, but trippable, rod until the next reactor shutdown (MODE 3 entry). If more than one rod cannot be moved, Condition 3.1.4 E must be entered. The plant must be in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

The analyses do not account for the possibility of more then one rod failing to insert on a trip. While boron concentration might be adjusted to restore SHUTDOWN MARGIN, if two adjacent rods fail to insert that portion of the core could remain excessively reactive. Since the analyses must assume that one rod fails to insert, operation may not continue with a known untrippable rod. A shutdown would be required by Condition 3.1.4 E.

A.1

Rod position indication is required to allow verification that the rods are positioned and aligned as assumed in the safety analysis. If one rod position indication channel is inoperable for one or more control rods then SR 3.1.4.1 (rod position verification) is required to be performed once within 15 minutes following any rod motion in that group. This ensures that the rods are positioned as required.

BASES

ACTIONS (continued)

B.1

When the control rod deviation alarm is inoperable, performing SR 3.1.4.1, once within 15 minutes of movement of any control rod, ensures improper control rod alignments are identified before unacceptable flux distributions occur. The specified Completion Times take into account other information continuously available to the operator in the control room, so that during control rod movement, deviations can be detected, and the protection provided by the control rod and deviation circuit is not required.

C.1 and C.2

Condition C addresses the situation where one rod in a group is misaligned, ie. there is more than 8 inches between that rod and any other rod in its group, but all remaining rods in that group are within 8 inches of each other.

A full-length control rod may become misaligned yet remain trippable. In this condition, the control rod can still perform its required function of adding negative reactivity should a reactor trip be necessary.

Regulating rod alignment can be restored by either aligning the misaligned rod(s) to within 8 inches of all other rods in its group or, aligning the misaligned rod's group to within 8 inches of the misaligned rod if allowed by the rod group insertion limits. Shutdown rod alignment can be restored by aligning the misaligned rod to within 8 inches of all other rods in its group.

If one control rod is misaligned by > 8 inches continued operation in MODES 1 and 2 may continue, provided, within 2 hours, the TOTAL RADIAL PEAKING FACTOR has been verified acceptable in accordance with SR 3.2.2.1, or the power is reduced to ≤ 75% RTP.

Xenon redistribution in the core starts to occur as soon as a rod becomes misaligned. Reducing THERMAL POWER to ≤ 75% RTP ensures acceptable power distributions are maintained.

BASES

ACTIONS

C.1 and C.2 (continued)

For small misalignments of the control rods, there is:

a. A small effect on the time dependent long-term power distributions relative to those used in generating LCOs and Limiting Safety System Settings (LSSS) setpoints;

b. A negligible effect on the available SDM; and

c. A small effect on the ejected rod worth used in the accident analysis.

With a large control rod misalignment, however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time dependent, long-term power distributions relative to those used in generating LCOs and LSSS setpoints.

The effect on the available SDM and the ejected rod worth used in the accident analysis remains small.

In both cases, a 2-hour time period is sufficient to:

a. Identify cause of a misaligned rod;

b. Take appropriate corrective action to realign the rods; and

c. Minimize the effects of xenon redistribution.

The Palisades analysis for rod misalignment is bounded by a single dropped rod. Therefore, rod misalignments are limited to one rod being misaligned from its group. If a full-length control rod is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable full-length control rod, meeting the insertion limits of LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and LCO 3.1.6, "Regulating Rod Group Position Limits," does not ensure that adequate SDM exists and therefore, the Actions of Condition E must be met.

BASES

ACTIONS (continued)

D.1

Condition D is entered whenever it is discovered that a single full-length control rod cannot be moved by its operator, yet the control rod is still capable of being tripped (or is fully inserted). Although the ability to move a full-length control rod is not an initial assumption used in the safety analyses, it does relate to full-length control rod OPERABILITY. The inability to move a full-length control rod by its operator may be indicative of a systemic failure (other than trippability) that could potentially affect other rods. Thus, declaring a full-length control rod inoperable in this instance is conservative since it limits the number of full-length control rods that cannot be moved by their operators to only one. The Completion Time to restore an inoperable control rod to OPERABLE status is stated as prior to entering MODE 2 following next MODE 3 entry. This Completion Time allows unrestricted operation in MODES 1 and 2 while conservatively preventing a reactor startup with an immovable full-length control rod.

E.1

If the Required Action or associated Completion Time of Condition A, Condition B, Condition C, or Condition D is not met; one or more control rods are inoperable for reasons other than Condition D (ie, one full length control rod is inoperable for reasons other than being "immovable but trippable," or more than one control rod, whether full length or part length, are inoperable for any reasons); or two or more control rods are misaligned by > 8 inches, or two channels of control rod position indication are inoperable for one or more control rods, the plant is required to be brought to MODE 3. By being brought to MODE 3, the plant is brought outside its MODE of applicability. Continued operation is not allowed in the case of more than one control rod misaligned from any other rod in its group by > 8 inches, or two or more rods inoperable. This is because these cases may be indicative of a loss of SDM and power re-distribution, and a loss of safety function, respectively.

Also, if no rod position indication exists for one or more control rods, continued operation is not allowed because the safety analysis assumptions of rod position cannot be ensured.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual control rod positions are within 8 inches of all other control rods in the group at a 12-hour Frequency allows the operator to detect a control rod that is beginning to deviate from its expected position. The specified Frequency takes into account other control rod position information that is continuously available to the operator in the control room, so that during control rod movement, deviations can be detected. Also protection can be provided by the control rod deviation alarm.

SR 3.1.4.2

OPERABILITY of two control rod position indicator channels is required to determine control rod positions, and thereby ensure compliance with the control rod alignment and insertion limits. Performance of a CHANNEL CHECK on the primary and secondary control rod position indication channels provides confidence in the accuracy of the rod position indication systems. The control rod "full in" and "full out" lights, which correspond to the lower electrical limit and the upper electrical limit respectively, provide an additional means for determining the control rod positions when the control rods are at either their fully inserted or fully withdrawn positions.

The 12-hour Frequency takes into consideration other information continuously available to the operator in the control room, so that during control rod movement, deviations can be detected, and protection can be provided by the control rod deviation alarm.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verifying each full-length control rod is trippable would require that each full-length control rod be tripped. In MODES 1 and 2, tripping each full-length control rod would result in radial or axial power tilts, or oscillations. Therefore, individual full-length control rods are exercised every 92 days to provide increased confidence that all full-length control rods continue to be trippable, even if they are not regularly tripped. A movement of 6 inches is adequate to demonstrate motion without exceeding the alignment limit when only one control rod is being moved. The 92-day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the control rods. At any time, if a control rod(s) is immovable, a determination of the trippability of the control rod(s) must be made, and appropriate action taken. Condition 3.1.4 D would apply whenever it is discovered that a single full-length control rod cannot be moved by its operator, yet the control rod is still capable of being tripped (or is fully inserted.)

SR 3.1.4.4

Demonstrating the rod position deviation alarm is OPERABLE verifies the alarm is functional. The 18-month Frequency takes into account other information continuously available to the operator in the control room, so that during control rod movement, deviations can be detected.

SR 3.1.4.5

Performance of a CHANNEL CALIBRATION of each control rod position indication channel ensures the channel is OPERABLE and capable of indicating control rod position over the entire length of the control rod's travel with the exception of the secondary rod position indicating channel dead band near the bottom of travel. This dead band exists because the control rod drive mechanism housing seismic support prevents operation of the reed switches. Since this Surveillance must be performed when the reactor is shut down, an 18-month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other surveillances being performed at shorter Frequencies, which determine the OPERABILITY of the control rod position indicating systems.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.6

Verification of full-length control rod drop times determines that the maximum control rod drop time is consistent with the assumed drop time used in that safety analysis (Ref. 2). The 2.5-second acceptance criteria is measured from the time the CRDM clutch is deenergized by the reactor protection system or test switch to 90% insertion. This time is bounded by that assumed in the safety analysis (Ref.2). Measuring drop times prior to reactor criticality, after reactor vessel head reinstallation, ensures that reactor internals and CRDMs will not interfere with full-length control rod motion or drop time and that no degradation in these systems has occurred that would adversely affect full-length control rod motion or drop time. Individual full-length control rods whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality, based on the need to perform this Surveillance under the conditions that apply during a plant outage and because of the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.1
3. FSAR, Section 14.4
4. FSAR, Section 14.6

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Special Test Exceptions (STE)

BASES

BACKGROUND

The primary purpose of this STE is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine control rod worths, SHUTDOWN MARGIN (SDM), and specific reactor core characteristics.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, tests, and experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in design and analyses;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required during startup and low power operation after each shutdown that involved an alteration of the fuel assemblies in the reactor core. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed.

BASES

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved in accordance with the administrative processes for procedure controls. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to power escalation.

Examples of PHYSICS TESTS include determination of critical boron concentration, full-length control rod group and individual control rod worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during a PHYSICS TEST with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-2005 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the Linear Heat Rate (LHR) remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

a. LCO 3.1.4, "Control Rod Alignment";

b. LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits";

c. LCO 3.1.6, "Regulating Rod Group Position Limits"; and

d. LCO 3.4.2, "PCS Minimum Temperature for Criticality."

This STE places limits on allowable THERMAL POWER during PHYSICS TESTS assuring the LHR and the Departure from Nucleate Boiling (DNB) parameters will be maintained within limits. It also places limits on the amount of control rod worth required to be available for reactivity control when control rod worth measurements are performed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

SRs are conducted as necessary to ensure that reactor power and shutdown capability remain within limits during PHYSICS TESTS. Requiring ≥ 1% shutdown reactivity, based on predicted control rod worths, be available for trip insertion from the OPERABLE full-length control rod provides a high degree of assurance that shutdown capability is maintained for the most challenging postulated accident assuming all full-length control rods are inserted in the core. Since LCOs 3.1.5 and 3.1.6 are suspended, however, there is not the same degree of assurance during this test that the reactor would always be shut down if the highest worth full-length control rod was stuck out and calculational uncertainties or the estimated highest rod worth was not as expected (the single failure criterion is not met). This situation is judged acceptable, however, because specified acceptable fuel damage limits are still met. The risk of experiencing a stuck rod and subsequent criticality is reduced during this PHYSICS TEST exception by the requirement that ≥ 1% shutdown reactivity is available based on predicted control rod worths.

PHYSICS TESTS include measurement of core parameters or exercise of control components. Also involved are the shutdown and regulating rods, which affect power peaking and are required for shutdown of the reactor. The limits for insertion of these rod groups are specified for each fuel cycle in the COLR.

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

LCO

This LCO relaxes the minimum primary coolant temperature at which the reactor may be made critical, permits individual full-length control rods and full-length control rod groups to be positioned outside of their normal alignment and insertion limits during the performance of PHYSICS TESTS such as those required to:

a. Measure control rod worths;

b. Measure control rod shadowing factors; and

c. Measure temperature and power coefficients.

BASES

LCO (continued)

This LCO specifies that a minimum amount of rod worth is immediately available for reactivity control when rod worth measurement tests are performed. This portion of the STE permits the periodic verification of the actual versus predicted control rod group worths.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS, provided:

a. THERMAL POWER is ≤ 2% RTP;

b. ≥ 1% shutdown reactivity, based on predicted control rod worth, is available for trip insertion; and

c. T_{ave} is ≥ 500°F.

APPLICABILITY

This LCO is applicable in MODE 2 because the reactor must be critical to perform the PHYSICS TESTS described in the LCO section.

ACTIONS

A.1

If THERMAL POWER exceeds 2% RTP, THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduction. The 15 minute Completion Time ensures that prompt action shall be taken to reduce THERMAL POWER to within acceptable limits.

B.1

If < 1% shutdown reactivity is available for trip insertion, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until ≥ 1% shutdown reactivity is achieved.

C.1

If the T_{ave} requirement is not met, T_{ave} must be restored. The 15 minutes Completion Time ensures that prompt action shall be taken to raise T_{ave} within the required limit.

BASES

ACTIONS (continued)

D.1

If Required Actions of Condition A, Condition B, or Condition C cannot be completed within the required Completion Time, PHYSICS TESTS must be suspended within 1 hour. Allowing 1 hour for suspending PHYSICS TESTS allows the operator sufficient time to change any abnormal rod configuration back to within the limits of LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6, or to restore Primary Coolant System (PCS) temperature to within the limits of LCO 3.4.2.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verifying that THERMAL POWER is ≤ 2% RTP as specified in the PHYSICS TEST procedure and required by the safety analysis, ensures that adequate LHR and DNB parameter margins are maintained while LCOs are suspended. The 1 hour Frequency is sufficient, based on the slow rate of power change and increased operational controls in place during PHYSICS TESTS.

SR 3.1.7.2

Verifying $T_{ave} \geq 500°F$ during the PHYSICS TEST ensures that T_{ave} remains in an analyzed range while the LCOs are suspended. The 1 hour Frequency is sufficient, based on the slow rate of change and increased operational controls in place during PHYSICS TESTS.

SR 3.1.7.3

Verification that ≥ 1% shutdown reactivity is available for trip insertion is performed by a reactivity balance calculation, considering the following reactivity effects:

a. PCS boron concentration;

b. Control rod group position;

c. PCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.3 (continued)

e. Xenon concentration; and

f. Isothermal Temperature Coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because reactor power is maintained below 2% RTP, and for most of the PHYSIC TESTS below the point of adding heat the fuel temperature will be changing at the same rate as the PCS.

The Frequency of 24 hours is based on the generally slow change in boron concentration and on the low probability of an accident occurring without the SDM established by LCO 3.1.5.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI

2. 10 CFR 50.59

3. Regulatory Guide 1.68, Revision 2, August 1978

4. ANSI/ANS-19.6.1-2005, November 29, 2005

B 3.3 INSTRUMENTATION

B 3.3.3 Engineered Safety Features (ESF) Instrumentation

BASES

BACKGROUND

The ESF Instrumentation initiates necessary safety systems, based upon the values of selected plant parameters, to protect against violating core design limits and the Primary Coolant System (PCS) pressure boundary and to mitigate accidents.

The ESF circuitry generates the signals listed below when the monitored variables reach levels that are indicative of conditions requiring protective action. The inputs to each ESF actuation signal are also listed.

1. Safety Injection Signal (SIS).
   a. Containment High Pressure (CHP)
   b. Pressurizer Low Pressure
2. Steam Generator Low Pressure (SGLP);
   a. Steam Generator A Low Pressure
   b. Steam Generator B Low Pressure
3. Recirculation Actuation Signal (RAS);
   a. Safety Injection Refueling Water Tank (SIRWT) Low Level
4. Auxiliary Feedwater Actuation Signal (AFAS);
   a. Steam Generator A Low Level
   b. Steam Generator B Low Level
5. Containment High Pressure Signal (CHP);
   a. Containment High Pressure - Left Train
   b. Containment High Pressure - Right Train

BASES

BACKGROUND (continued)

6. Containment High Radiation Signal (CHR);

   a. Containment High Radiation

7. Automatic Bypass Removal

   a. Pressurizer Pressure Low Bypass

   b. Steam Generator A Low Pressure Bypass

   c. Steam Generator B Low Pressure Bypass

In the above list of actuation signals, the CHP and RAS are derived from pressure and level switches, respectively.

Equipment actuated by each of the above signals is identified in the FSAR, Chapter 7. (Ref. 1).

The ESF circuitry, with the exception of RAS, employs two-out-of-four logic. Four independent measurement channels are provided for each function used to generate ESF actuation signals. When any two channels of the same function reach their setpoint, actuating relays are energized which, in turn, initiate the protective actions. Two separate and redundant trains of actuating relays, each powered from separate power supplies, are utilized. These separate relay trains operate redundant trains of ESF equipment.

RAS logic consists of output contacts of the relays actuated by the SIRWT level switches arranged in a "one-out-of-two taken twice" logic. The contacts are arranged so that at least one low level signal powered from each station battery is required to initiate RAS. Loss of a single battery, therefore, cannot either cause or prevent RAS initiation.

The ESF logic circuitry contains the capability to manually block the SIS actuation logic and the SGLP action logic during normal plant shutdowns to avoid undesired actuation of the associated equipment. In each case, when three of the four associated measurement channels are below the block setpoint, pressing a manual pushbutton will block the actuation signal for that train. If two of the four of the measurement channels increase above the block setpoint, the block will automatically be removed.

BASES

BACKGROUND
(continued)

The sensor subsystems, including individual channel actuation bistables, is addressed in this LCO. The actuation logic subsystems, manual actuation, and downstream components used to actuate the individual ESF components are addressed in LCO 3.3.4.

Measurement Channels

Measurement channels, consisting of pressure switches, field transmitters, or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

Four identical measurement channels are provided for each parameter used in the generation of trip signals. These are designated Channels A through D. Measurement channels provide input to ESF bistables within the same ESF channel. In addition, some measurement channels may also be used as inputs to Reactor Protective System (RPS) bistables, and most provide indication in the control room.

When a channel monitoring a parameter indicates an abnormal condition, the bistable monitoring the parameter in that channel will trip. In the case of RAS and CHP, the sensors are latching auxiliary relays from level and pressure switches, respectively, which do not develop an analog input to separate bistables. Tripping two or more channels monitoring the same parameter will actuate both channels of Actuation Logic of the associated ESF equipment.

Three of the four measurement and bistable channels are necessary to meet the redundancy and testability of GDC 21 in Appendix A to 10 CFR 50 (Ref. 2). The fourth channel provides additional flexibility by allowing one channel to be removed from service for maintenance or testing while still maintaining a minimum two-out-of-three logic.

Since no single failure will prevent a protective system actuation and no protective channel feeds a control channel, this arrangement meets the requirements of IEEE Standard 279 -1971 (Ref. 3).

BASES

BACKGROUND (continued)

Measurement Channels (continued)

The ESF Actuation Functions are generated by comparing a single measurement to a fixed bistable setpoint. The ESF Actuation Functions utilize the following input instrumentation:

- Safety Injection Signal (SIS)

  The Safety Injection Signal can be generated by any of three inputs: Pressurizer Low Pressure, Containment High Pressure, or Manual Actuation. Manual Actuation is addressed by LCO 3.3.4; Containment High Pressure is discussed below. Four instruments (channels A through D), monitor Pressurizer Pressure to develop the SIS actuation. Each of these instrument channels has two individually adjustable ESF bistable trip devices, one for the bypass removal circuit (discussed below) and one for SIS. Each ESF bistable trip device actuates two auxiliary relays, one for each actuation train. The output contacts from these auxiliary relays form the logic circuits addressed in LCO 3.3.4. The instrument channels associated with each Pressurizer Low Pressure SIS actuation bistable include the pressure measurement loop, the SIS actuation bistable, and the two auxiliary relays associated with that bistable. The bistables associated with automatic removal of the Pressurizer Low Pressure Bypass are discussed under Function 7.a, below.

- Low Steam Generator Pressure Signal (SGLP)

  There are two separate Low Steam Generator Pressure signals, one for each steam generator. For each steam generator, four instruments (channels A through D) monitor pressure to develop the SGLP actuation. Each of these instrument channels has two individually adjustable ESF bistable trip devices, one for the bypass removal circuit (discussed below) and one for SGLP. Each Steam SGLP bistable trip device actuates an auxiliary relay. The output contacts from these auxiliary relays form the SGLP logic circuits addressed in LCO 3.3.4. The instrument channels associated with each Steam Generator Low Pressure Signal bistable include the pressure measurement loop, the SGLP actuation bistable, and the auxiliary relay associated with that bistable. The bistables associated with automatic removal of the SGLP Bypass are discussed under Function 7.a, below.

BASES

BACKGROUND (continued)

Measurement Channels (continued)

- Recirculation Actuation Signal (RAS)

  There are four Safety Injection Refueling Water (SIRW) Tank level instruments used to develop the RAS signal. Each of these instrument channels actuates two auxiliary relays, one for each actuation train. The output contacts from these auxiliary relays form the logic circuits addressed in LCO 3.3.4. The SIRW Tank Low Level instrument channels associated with each RAS actuation bistable include the level instrument and the two auxiliary relays associated with that instrument.

- Auxiliary Feedwater Actuation Signal (AFAS)

  There are two separate AFAS signals (AFAS channels A and B), each one actuated on low level in either steam generator. For each steam generator, four level instruments (channels A through D) monitor level to develop the AFAS actuation signals. The output contacts from the bistables on these level channels form the AFAS logic circuits addressed in LCO 3.3.4. The instrument channels associated with each Steam Generator Low Level Signal bistable include the level measurement loop and the Low Level AFAS bistable.

- Containment High Pressure Actuation (CHP)

  The Containment High Pressure signal is actuated by two sets of four pressure switches, one set for each train. The output contacts from these pressure switches form the CHP logic circuits addressed in LCO 3.3.4.

BASES

BACKGROUND (continued)

Measurement Channels (continued)

- Containment High Radiation Actuation (CHR)

  The CHR signal can be generated by either of two inputs: High Radiation or Manual Actuation. Manual Actuation is addressed by LCO 3.3.4. Four radiation monitor instruments (channels A through D), monitor containment area radiation level to develop the CHR signal. Each CHR monitor bistable device actuates one auxiliary relay which has contacts in each CHR logic train addressed in LCO 3.3.4. The instrument channels associated with each CHR actuation bistable include the radiation monitor itself and the associated auxiliary relay.

- Automatic Bypass Removal Functions

  Pressurizer Low Pressure and Steam Generator Low Pressure logic circuits have the capability to be blocked to avoid undesired actuation when pressure is intentionally lowered during plant shutdowns. In each case these bypasses are automatically removed when the measured pressure exceeds the bypass permissive setpoint. The measurement channels which provide the bypass removal signal are the same channels which provide the actuation signal. Each of these pressure measurement channels has two bistables, one for actuation and one for the bypass removal Function. The pressurizer pressure channels include an auxiliary relay actuated by the bypass removal bistable. The logic circuits for Automatic Bypass Removal Functions are addressed by LCO 3.3.4.

Several measurement instrument channels provide more than one required function. Those sensors shared for RPS and ESF functions are identified in Table B 3.3.1-1. That table provides a listing of those shared channels and the Specifications which they affect.

BASES

BACKGROUND (continued)

Bistable Trip Units

There are four channels of bistables, designated A through D, for each ESF Function, one for each measurement channel. The bistables for all required Functions, except CHP and RAS, receive an analog input from the measurement device, compare the analog input to trip setpoints, and provide contact output to the Actuation Logic. CHP and RAS are actuated by pressure switches and level switches respectively.

The Allowable Values are specified for each safety related ESF trip Function which is credited in the safety analysis. Nominal trip setpoints are specified in the plant procedures. The nominal setpoints are selected to ensure plant parameters do not exceed the Allowable Value if the instrument loop is performing as required. The methodology used to determine the nominal trip setpoints is also provided in plant documents. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified is more conservative than the analytical limit determined in the safety analysis in order to account for uncertainties appropriate to the trip Function. These uncertainties are addressed as described in plant documents. A channel is inoperable if its actual setpoint is not within its Allowable Value.

Setpoints in accordance with the Allowable Value will ensure that Safety Limits of Chapter 2.0, "SAFETY LIMITS (SLs)," are not violated during Anticipated Operational Occurrences (AOOs) and that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed. (As defined in 10 CFR 50, Appendix A, "Anticipated operational occurances mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.")

ESF Instrument Channel Bypasses

The only ESF instrument channels with built-in bypass capability are the Low SG Level AFAS bistables. Those bypasses are effected by a key operated switch, similar to the RPS Trip Channel Bypasses. A bypassed Low SG Level channel AFAS bistable cannot perform its specified function and must be considered inoperable.

BASES

BACKGROUND (continued)

ESF Instrument Channel Bypasses (continued)

While there are no other built-in provisions for instrument channel bypasses in the ESF design (bypassing any other channel output requires opening a circuit link, lifting a lead, or using a jumper), this LCO includes requirements for OPERABILITY of the instrument channels and bistables which provide input to the Automatic Bypass Removal Logic channels required by LCO 3.3.4, "ESF Logic and Manual Initiation."

The Actuation Logic channels for Pressurizer Pressure and Steam Generator Low Pressure, however, have the ability to be manually bypassed when the associated pressure is below the range where automatic protection is required. These actuation logic channel bypasses may be manually initiated when three-out-of-four bypass permissive bistables indicate below their setpoint. When two-out-of-four of these bistables are above their bypass permissive setpoint, the actuation logic channel bypass is automatically removed. The bypass permissive bistables use the same four measurement channels as the blocked ESF function for their inputs.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESF Functions. One of the ESF Functions is the primary actuation signal for that accident. An ESF Function may be the primary actuation signal for more than one type of accident. An ESF Function may also be a secondary, or backup, actuation signal for one or more other accidents. Functions not specifically credited in the accident analysis, serve as backups and are part of the NRC approved licensing basis for the plant.

BASES

APPLICABLE SAFETY ANALYSES (continued)

ESF protective Functions are as follows.

1. Safety Injection Signal (SIS)

The SIS ensures acceptable consequences during Loss of Coolant Accident (LOCA) events, including steam generator tube rupture, and Main Steam Line Breaks (MSLBs) or Feedwater Line Breaks (FWLBs) (inside containment). To provide the required protection, SIS is actuated by a CHP signal, or by two-out-of-four Pressurizer Low Pressure channels decreasing below the setpoint. SIS initiates the following actions:

a. Start HPSI & LPSI pumps;

b. Start component cooling water and service water pumps;

c. Initiate service water valve operations;

d. Initiate component cooling water valve operations;

e. Start containment cooling fans (when coincident with a loss of offsite power);

f. Enable Containment Spray Pump Start on CHP; and

g. Initiate Safety Injection Valve operations.

Each SIS logic train is also actuated by a contact pair on one of the CHP initiation relays for the associated CHP train.

2. Steam Generator Low Pressure Signal (SGLP)

The SGLP ensures acceptable consequences during an MSLB or FWLB by isolating the steam generator if it indicates a low steam generator pressure. The SGLP concurrent with or following a reactor trip, minimizes the rate of heat extraction and subsequent cooldown of the PCS during these events.

BASES

APPLICABLE SAFETY ANALYSES (continued)

2. Steam Generator Low Pressure Signal (SGLP) (continued)

One SGLP circuit is provided for each SG. Each SGLP circuit is actuated by two-out-of-four pressure channels on the associated SG reaching their setpoint. SGLP initiates the following actions:

a. Close the associated Feedwater Regulating valve and its bypass; and

b. Close both Main Steam Isolation Valves.

3. Recirculation Actuation Signal

At the end of the injection phase of a LOCA, the SIRWT will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from SIRWT to the containment sump must occur before the SIRWT empties to prevent damage to the ECCS pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support pump suction.

Furthermore, early switchover must not occur to ensure sufficient borated water is injected from the SIRWT to ensure the reactor remains shut down in the recirculation mode. An SIRWT Low Level signal initiates the RAS.

RAS initiates the following actions:

a. Trip LPSI pumps (this trip can be manually bypassed);

b. Switch HPSI and containment spray pump suction from SIRWT to Containment Sump by opening sump CVs and closing SIRWT CVs;

c. Adjust cooling water to component cooling heat exchangers;

d. Open HPSI subcooling valve CV-3071 if the associated HPSI pump is operating;

e. After containment sump valve CV-3030 is opened, open HPSI subcooling valve CV-3070 if the associated HPSI pump is operating;

f. Close containment spray valve CV-3001 if containment sump valve CV-3030 does not open.

BASES

APPLICABLE SAFETY ANALYSES (continued)

3 Recirculation Actuation Signal (continued)

The RAS signal is actuated by separate sensors from those which provide tank level indication. The allowable range of 21" to 27" above the tank floor corresponds to 1.1% to 3.3% indicated level. Typically the actual setting is near the midpoint of the allowable range.

4 Auxiliary Feedwater Actuation Signal

An AFAS initiates feedwater flow to both steam generators if a low level is indicated in either steam generator.

The AFAS maintains a steam generator heat sink during the following events:

- MSLB;
- FWLB;
- LOCA; and
- Loss of feedwater.

5. Containment High Pressure Signal (CHP)

The CHP signal closes all containment isolation valves not required for ESF operation and starts containment spray (if SIS enabled), ensuring acceptable consequences during LOCAs, control rod ejection events, MSLBs, or FWLBs (inside containment).

CHP is actuated by two-out-of-four pressure switches for the associated train reaching their setpoints. CHP initiates the following actions:

a. Containment Spray;

b. Safety Injection Signal;

c. Main Feedwater Isolation;

BASES

APPLICABLE SAFETY ANALYSIS (continued)

5. Containment High Pressure Signal (CHP) (continued)

   d. Main Steam Line Isolation;

   e. Control Room HVAC Emergency Mode; and

   f. Containment Isolation Valve Closure.

6. Containment High Radiation Signal (CHR)

   CHR is actuated by two-out-of-four radiation monitors exceeding their setpoints. CHR initiates the following actions to ensure acceptable consequences following a LOCA or control rod ejection event:

   a. Control Room HVAC Emergency Mode;

   b. Containment Isolation Valve Closure; and

   c. Block automatic starting of ECCS pump room sump pumps.

   During refueling operations, separate switch-selectable radiation monitors initiate CHR, as addressed by LCO 3.3.6.

7. Automatic Bypass Removal Functions

   The logic circuitry provides automatic removal of the Pressurizer Pressure Low and Steam Generator Pressure Low actuation signal bypasses. There are no assumptions in the safety analyses which assume operation of these automatic bypass removal circuits, and no analyzed events result in conditions where the automatic removal would be required to mitigate the event. The automatic removal circuits are required to assure that logic circuit bypasses will not be overlooked during a plant startup.

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

BASES

LCO

The LCO requires all channel components necessary to provide an ESF actuation to be OPERABLE.

The Bases for the LCO on ESF Functions are addressed below.

1. Safety Injection Signal (SIS)

   This LCO requires four channels of SIS Pressurizer Low Pressure to be OPERABLE in MODES 1, 2, and 3.

   The setpoint was chosen so as to be low enough to avoid actuation during plant operating transients, but to be high enough to be quickly actuated by a LOCA or MSLB. The settings include an uncertainty allowance which is consistent with the settings assumed in the MSLB analysis (which bounds the settings assumed in the LOCA analysis).

2. Steam Generator Low Pressure Signal (SGLP)

   This LCO requires four channels of Steam Generator Low Pressure Instrumentation for each SG to be OPERABLE in MODES 1, 2, and 3. However, as indicated in Table 3.3.3-1, Note (a), the SGLP Function is not required to be OPERABLE in MODES 2 or 3 if all Main Steam Isolation Valves (MSIVs) are closed and deactivated and all Main Feedwater Regulating Valves (MFRVs) and MFRV bypass valves are either closed and deactivated or isolated by closed manual valves.

   The setpoint was chosen to be low enough to avoid actuation during plant operation, but be close enough to full power operating pressure to be actuated quickly in the event of a MSLB. The setting includes an uncertainty allowance which is consistent with the setting used in the Reference 4 analysis.

   Each SGLP logic is made up of output contacts from four pressure bistables from the associated SG. When the logic circuit is satisfied, two relays are energized to actuate steam and feedwater line isolation.

BASES

LCO (continued)

2. Steam Generator Low Pressure Signal (SGLP) (continued)

This LCO applies to failures in the four sensor subsystems, including sensors, bistables, and associated equipment. Failures in the actuation subsystems are considered Actuation Logic failures and are addressed in LCO 3.3.4.

3. Recirculation Actuation Signal (RAS)

This LCO requires four channels of SIRWT Low Level to be OPERABLE in MODES 1, 2, and 3.

The setpoint was chosen to provide adequate water in the containment sump for HPSI pump net positive suction head following an accident, but prevent the pumps from running dry during the switchover.

The upper limit on the Allowable Value for this trip is set low enough to ensure RAS does not initiate before sufficient water is transferred to the containment sump. Premature recirculation could impair the reactivity control Function of safety injection by limiting the amount of boron injection. Premature recirculation could also damage or disable the recirculation system if recirculation begins before the sump has enough water.

The lower limit on the SIRWT Low Level trip Allowable Value is high enough to transfer suction to the containment sump prior to emptying the SIRWT.

4. Auxiliary Feedwater Actuation Signal (AFAS)

The AFAS logic actuates AFW to each SG on a SG Low Level in either SG.

The Allowable Value was chosen to assure that AFW flow would be initiated while the SG could still act as a heat sink and steam source, and to assure that a reactor trip would not occur on low level without the actuation of AFW.

BASES

LCO (continued)

4. Auxiliary Feedwater Actuation Signal (AFAS) (continued)

This LCO requires four channels for each steam generator of Steam Generator Low Level to be OPERABLE in MODES 1, 2, and 3.

5. Containment High Pressure Signal (CHP)

This LCO requires four channels of CHP to be OPERABLE for each of the associated ESF trains (left and right) in MODES 1, 2, 3 and 4.

The setpoint was chosen so as to be high enough to avoid actuation by containment temperature or atmospheric pressure changes, but low enough to be quickly actuated by a LOCA or a MSLB in the containment.

6. Containment High Radiation Signal (CHR)

This LCO requires four channels of CHR to be OPERABLE in MODES 1, 2, 3, and 4.

The setpoint is based on the maximum primary coolant leakage to the containment atmosphere allowed by LCO 3.4.13 and the maximum activity allowed by LCO 3.4.16. N^{16} concentration reaches equilibrium in containment atmosphere due to its short half-life, but other activity was assumed to build up. At the end of a 24 hour leakage period the dose rate is approximately 20 R/h as seen by the area monitors. A large leak could cause the area dose rate to quickly exceed the 20 R/h setting and initiate CHR.

7. Automatic Bypass Removal

The automatic bypass removal logic removes the bypasses which are used during plant shutdown periods, for Pressurizer Low Pressure and Steam Generator Low Pressure actuation signals.

The setpoints were chosen to be above the setpoint for the associated actuation signal, but well below the normal operating pressures.

BASES

LCO (continued)

7. Automatic Bypass Removal (continued)

This LCO requires four channels of Pressurizer Low Pressure bypass removal and four channels for each steam generator of Steam Generator Low Pressure bypass removal, to be OPERABLE in MODES 1, 2, and 3.

APPLICABILITY

All ESF Functions are required to be OPERABLE in MODES 1, 2, and 3. In addition, Containment High Pressure and Containment High Radiation are required to be operable in MODE 4.

In MODES 1, 2, and 3 there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the main steam isolation valves to preclude a positive reactivity addition and containment overpressure;
- Actuate AFW to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available);
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or MSLB; and
- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

The CHP and CHR Functions are required to be OPERABLE in MODE 4 to limit leakage of radioactive material from containment and limit operator exposure during and following a DBA.

The SGLP Function is not required to be OPERABLE in MODES 2 and 3, if all MSIVs are closed and deactivated and all MFRVs and MFRV bypass valves are either closed and deactivated or isolated by closed manual valves, since the SGLP Function is not required to perform any safety functions under these conditions.

BASES

APPLICABILITY (continued)

In lower MODES, automatic actuation of ESF Functions is not required, because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating the ESF components.

LCO 3.3.6 addresses automatic Refueling CHR isolation during CORE ALTERATIONS or during movement of irradiated fuel.

In MODES 5 and 6, ESFAS initiated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components.

ACTIONS

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

Typically, the drift is small and results in a delay of actuation rather than a total loss of function. Determination of setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the actual trip setpoint is not within the Allowable Value in Table 3.3.3-1, the channel is inoperable and the appropriate Condition(s) are entered.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value in Table 3.3.3-1, or the sensor, instrument loop, signal processing electronics, or ESF bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the plant must enter the Condition statement for the particular protection Function affected.

BASES

ACTIONS (continued)

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A Note has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.3-1. Completion Times for the inoperable channel of a Function will be tracked separately.

A.1

Condition A applies to the failure of a single bistable or associated instrumentation channel of one or more input parameters in each ESF Function except the RAS Function. Since the bistable and associated instrument channel combine to perform the actuation function, the Condition is also appropriate if both the bistable and associated instrument channel are inoperable.

ESF coincidence logic is normally two-out-of-four. If one ESF channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one ESF channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel actuation bistable is placed in trip within 7 days. The provision of four trip channels allows one channel to be inoperable in a non-trip condition up to the 7 day Completion Time allotted to place the channel in trip. Operating with one failed channel in a non-trip condition during operations, places the ESF Actuation Logic in a two-out-of-three coincidence logic.

If the failed channel cannot be restored to OPERABLE status in 7 days, the associated bistable is placed in a tripped condition. This places the function in a one-out-of-three configuration.

BASES

ACTIONS (continued)

A.1 (continued)

In this configuration, common cause failure of the dependent channel cannot prevent ESF actuation. The 7 day Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 7 day period is a low probability event.

Condition A is modified by a Note which indicates it is not applicable to the SIRWT Low Level Function.

B.1 and B.2

Condition B applies to the failure of two channels in any of the ESF Functions except the RAS Function.

With two inoperable channels, one channel actuation device must be placed in trip within the 8 hour Completion Time. Eight hours is allowed for this action since it must be accomplished by a circuit modification, or by removing power from a circuit component. With one channel of protective instrumentation inoperable, the ESF Actuation Logic Function is in two-out-of-three logic, but with another channel inoperable the ESF may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESF in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, ESF actuation will occur.

One of the failed channels must be restored to OPERABLE status within 7 days, and the provisions of Condition A still applied to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 must be placed in trip if more than 7 days has elapsed since the channel's initial failure.

BASES

ACTIONS (continued)

B.1 and B.2 (continued)

Condition B is modified by a Note which indicates that it is not applicable to the SIRWT Low Level Function.

C.1 and C.2

Condition C applies to one RAS SIRWT Low Level channel inoperable. The SIRWT low level circuitry is arranged in a "1-out-of-2 taken twice" logic rather than the more frequently used 2-out-of-4 logic. Therefore, Required Action C.1 differs from other ESF functions. With a bypassed SIRWT low level channel, an additional failure might disable automatic RAS, but would not initiate a premature RAS. With a tripped channel, an additional failure could cause a premature RAS, but would not disable the automatic RAS.

Since considerable time is available after initiation of SIS until RAS must be initiated, and since a premature RAS could damage the ESF pumps, it is preferable to bypass an inoperable channel and risk loss of automatic RAS than to trip a channel and risk a premature RAS.

The Completion Time of 8 hours allowed is reasonable because the Required Action involves a circuit modification.

Required Action C.2 requires that the inoperable channel be restored to OPERABLE status within 7 days. The Completion Time is reasonable based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 7 day period is a low probability event.

BASES

ACTIONS (continued)

D.1 and D.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met for Functions 1, 2, 3, 4, or 7, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1 and E.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met for Functions 5 or 6, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for any particular ESF Function are found in the SRs column of Table 3.3.3-1 for that Function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.3.1

A CHANNEL CHECK is performed once every 12 hours on each ESF input channel which is provided with an indicator to provide a qualitative assurance that the channel is working properly and that its readings are within limits. A CHANNEL CHECK is not performed on the CHP and SIRWT Low Level channels because they have no associated control room indicator.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.1 (continued)

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.2

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

This test is required to be performed each 92 days on ESF input channels provided with on-line testing capability. It is not required for the SIRWT Low Level channels since they have no built in test capability. The CHANNEL FUNCTIONAL TEST for SIRWT Low Level channels is performed each 18 months as part of the required CHANNEL CALIBRATION.

The CHANNEL FUNCTIONAL TEST tests the individual channels using an analog test input to each bistable.

Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Reference 5).

SR 3.3.3.3

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.3 (continued)

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the extension analysis. The requirements for this review are outlined in Reference 5.

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

REFERENCES

1. FSAR, Chapter 7
2. 10 CFR 50, Appendix A
3. IEEE Standard 279-1971
4. FSAR, Chapter 14
5. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.4 PCS Loops - MODES 1 and 2

BASES

BACKGROUND

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

The secondary functions of the PCS include:

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

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

c. Carrying the soluble neutron poison, boric acid;

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

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

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

APPLICABLE SAFETY ANALYSES

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming four PCPs are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to PCP operation are the Loss of Forced Primary Coolant Flow, Primary Coolant Pump Rotor Seizure and Uncontrolled Control Rod Withdrawal events (Ref. 1).

Steady state DNB analysis had been performed for the four pump combination. The steady state DNB analysis, which generates the pressure and temperature and Safety Limit (i.e., the Departure from Nucleate Boiling Ratio (DNBR) limit), assumes a maximum power level of 110.4% RTP. This is the design overpower condition for four pump operation. The 110.4% value is the accident analysis setpoint of the trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

PCS Loops - MODES 1 and 2 satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2).

LCO

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

Each OPERABLE loop consists of two PCPs providing forced flow for heat transport to an SG that is OPERABLE. SG, and hence PCS loop OPERABILITY with regards to SG water level is ensured by the Reactor Protection System (RPS) in MODES 1 and 2. A reactor trip places the plant in MODE 3 if any SG water level is ≤ 25.9% (narrow range) as sensed by the RPS. The minimum level to declare the SG OPERABLE is 25.9% (narrow range).

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

BASES

APPLICABILITY

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, 5, and 6.

Operation in other MODES is covered by:

LCO 3.4.5, "PCS Loops-MODE 3";

LCO 3.4.6, "PCS Loops-MODE 4";

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

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

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

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

ACTIONS

A.1

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

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours of the required number of loops in operation. Verification may include indication of PCS flow, temperature, or pump status, which help to ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

REFERENCES

1. FSAR, Section 14.1

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.5 PCS Loops - MODE 3

BASES

BACKGROUND

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

In MODE 3, Primary Coolant Pumps (PCPs) are used to provide forced circulation heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single PCS loop with one PCP is sufficient to remove core decay heat. However, two PCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Any combination of OPERABLE PCPs and OPERABLE PCS loops can be used to fulfill the heat removal function.

Primary coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the PCS cannot be ensured. Any combination of OPERABLE PCPs and OPERABLE PCS loops can be used to fulfill the mixing function.

APPLICABLE SAFETY ANALYSES

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

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

LCO

The purpose of this LCO is to require two PCS loops to be available for heat removal, thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable (> -84% water level) of transferring heat from the primary coolant at a controlled rate. Forced primary coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running PCP meets the LCO requirement for one loop in operation.

BASES

LCO (continued)

Note 1 permits all PCPs to not be in operation ≤ 1 hour per 8 hour period. This means that natural circulation has been established using the SGs. The Note prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the PCS without the PCPs depends on the core decay heat load and the length of time that the pumps are stopped. As decay heat diminishes, the effects on PCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (Pressure and Temperature (P/T) limits or Low Temperature Overpressure Protection (LTOP) limits) must be observed and forced flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping the PCPs are to be limited to situations where:

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

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

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

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

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

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

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

BASES

LCO (continued)

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

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

An OPERABLE PCS loop consists of any one (of the four) OPERABLE PCP and an SG that is OPERABLE and has the minimum water level specified in SR 3.4.5.2. A PCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

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

Operation in other MODES is covered by:

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

LCO 3.4.6, "PCS Loops-MODE 4";

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

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

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

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

BASES (continued)

ACTIONS

A.1

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

B.1

If restoration is not possible within 72 hours, the plant must be placed in MODE 4 within 24 hours. In MODE 4, the plant may be placed on the SDC System. The Completion Time of 24 hours is compatible with required operation to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If no PCS loop is in operation, except as provided in Note 1 in the LCO section, all operations involving a reduction of PCS boron concentration must be immediately suspended. This is necessary because boron dilution requires forced circulation for proper homogenization. Action to restore one PCS loop to OPERABLE status and operation shall be initiated immediately and continued until one PCS loop is restored to OPERABLE status and operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required number of PCS loops are in operation. Verification include indication of PCS flow, temperature, and pump status, which help ensure that forced flow is providing heat removal and mixing of the soluble boric acid. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.5.2

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

SR 3.4.5.3

Verification that the required PCP is OPERABLE ensures that the single failure criterion is met and that an additional PCS loop can be placed in operation, if needed, to maintain decay heat removal and primary coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required PCP that is not in operation such that the PCP is capable of being started and providing forced PCS flow if needed. Proper breaker alignment and power availability means the breaker for the required PCP is racked-in and electrical power is available to energize the PCP motor. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES None

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.6 PCS Loops - MODE 4

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

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

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

LCO

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

BASES

LCO (continued)

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

Note 1 permits all PCPs and SDC pumps to not be in operation ≤ 1 hour per 8 hour period. The Note prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the PCS without the PCPs or SDC pumps depends on the core decay heat load and the length of time that the pumps are stopped. As decay heat diminishes, the effects on PCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the primary coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (Pressure and Temperature (P/T) limits or Low Temperature Overpressure Protection (LTOP) limits) must be observed and forced SDC flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both PCPs or SDC pumps are to be limited to situations where:

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

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

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

BASES

LCO (continued)

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

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

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

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

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

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

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

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. The two SDC heat exchangers operate as a single unit. A separate OPERABLE SDC heat exchanger is required for each OPERABLE SDC train. SDC pumps are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

BASES (continued)

APPLICABILITY

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

Operation in other MODES is covered by:

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

LCO 3.4.5, "PCS Loops-MODE 3";

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

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

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

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

ACTIONS

A.1

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

B.1

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

BASES

ACTIONS (continued)

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

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

SR 3.4.6.2

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional PCS loop or SDC train can be placed in operation, if needed to maintain decay heat removal and primary coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump that is not in operation such that the pump is capable of being started and providing forced PCS flow if needed. Proper breaker alignment and power availability means the breaker for the required pump is racked-in and electrical power is available to energize the pump motor. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.7 PCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

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

In MODE 5 with PCS loops filled, the SDC trains are the principal means for decay heat removal. The number of trains in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC train for decay heat removal and transport. The flow provided by one SDC train is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for decay heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an SDC train that must be OPERABLE and in operation. The second path can be another OPERABLE SDC train, or through the SGs, via natural circulation each having an adequate water level. "Loops filled" means the PCS loops are not blocked by dams and totally filled with coolant.

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

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

LCO

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

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

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

BASES

LCO (continued)

10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the PCS without the SDC pumps depends on the core decay heat load and the length of time that the pumps are stopped.

As decay heat diminishes, the effects on PCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the primary coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (Pressure and Temperature (P/T) limits or Low Temperature Overpressure Protection (LTOP) limits) must be observed and forced SDC flow or heat removal via the SGs must be re-established prior to reaching the pressure limit.

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

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

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

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

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

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

BASES

LCO (continued)

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

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

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

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. The two SDC heat exchangers operate as a single unit. A separate OPERABLE SDC heat exchanger is required for each OPERABLE SDC train. SDC pumps are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

An SG can perform as a heat sink via natural circulation when:

a. SG has the minimum water level specified in SR 3.4.7.2.

b. SG is OPERABLE.

c. SG has available method of feedwater addition and a controllable path for steam release.

d. Ability to pressurize and control pressure in the PCS.

If both SGs do not meet the above provisions, then LCO 3.4.7 item b (i.e. the secondary side water level of each SG shall be ≥ -84%) is not met.

BASES (continued)

APPLICABILITY

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

Operation in other MODES is covered by:

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

LCO 3.4.5, "PCS Loops-MODE 3";

LCO 3.4.6, "PCS Loops-MODE 4";

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

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

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

BASES (continued)

ACTIONS

A.1 and A.2

If one SDC train is inoperable and any SG has a secondary side water level < -84% (refer to LCO Bases section), redundancy for heat removal is lost. Action must be initiated immediately to restore a second SDC train to OPERABLE status or to restore the water level in the required SGs. Either Required Action A.1 or Required Action A.2 will restore redundant decay heat removal paths. The immediate Completion Times reflect the importance of maintaining the availability of two paths for decay heat removal.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.2

This SR requires verification every 12 hours of secondary side water level in the required SGs ≥ -84% using the wide range level instrumentation. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the primary coolant. The Surveillance is required to be performed when the LCO requirement is being met by use of the SGs. If both SDC trains are OPERABLE, this SR is not needed. The 12-hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify SG status.

SR 3.4.7.3

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

REFERENCES

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

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.8 PCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

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

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

BASES

LCO

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

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

Note 1 permits all SDC pumps to not be in operation for ≤ 1 hour. The Note prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a flow obstruction. Operations which could drain the PCS and thereby cause a loss of, or failure to regain SDC capability are also prohibited.

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

BASES

LCO (continued)

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

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. The two SDC heat exchangers operate as a single unit. A separate OPERABLE SDC heat exchanger is required for each OPERABLE SDC train. SDC pumps are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

APPLICABILITY

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

Operation in other MODES is covered by:

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

LCO 3.4.5, "PCS Loops-MODE 3";

LCO 3.4.6, "PCS Loops-MODE 4";

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

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

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

ACTIONS

A.1

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

BASES

ACTIONS (continued)

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1 and SR 3.4.8.2

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.8.3

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

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

SR 3.4.8.4

Verification that the required number of trains are OPERABLE ensures that redundant paths for heat removal are available and that additional trains can be placed in operation, if needed, to maintain decay heat removal and primary coolant circulation. Verification is performed by verifying proper breaker alignment and indicated power available to the required pump that is not in operation such that the SDC pump is capable of being started and providing forced PCS flow if needed. Proper breaker alignment and power availability means the breaker for the required SDC pump is racked-in and electrical power is available to energize the SDC pump motor. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES None

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.13 PCS Operational LEAKAGE

BASES

BACKGROUND

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

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

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

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

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

This LCO deals with protection of the Primary Coolant Pressure Boundary (PCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a Loss Of Coolant Accident (LOCA).

BASES

BACKGROUND (continued)

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

(1) Part of the primary coolant system, or

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

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

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

(iii) The pressurizer safety valves and PORVs.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for all events resulting in a discharge of steam from the steam generators to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 0.3 gpm or increases to 0.3 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR) and the Control Rod Ejection (CRE) accident analyses. The leakage contaminates the secondary fluid.

The FSAR (Ref. 2 and 5) analysis for SGTR assumes the contaminated secondary fluid is released via the Main Steam Safety Valves and Atmospheric Dump Valves. The 0.3 gpm primary to secondary LEAKAGE safety analysis assumption is inconsequential, relative to the dose contribution from the affected SG.

The MSLB (Ref 3 and 5) is more limiting than SGTR for site radiation releases. The safety analysis for the MSLB accident assumes the entire 0.3 gpm primary to secondary LEAKAGE is through the affected steam generator as an initial condition.

The CRE (Ref 4 and 5) accident with primary fluid release through the Atmospheric Dump Valves is the most limiting event for site radiation releases. The safety analysis for the CRE accident assumes 0.3 gpm primary to secondary LEAKAGE in one steam generator as an initial condition.

BASES

APPLICABLE SAFETY ANALYSES

The dose consequences resulting from the SGTR, MSLB and CRE accidents are well within the guidelines defined in 10 CFR 100 and meets the requirements of Appendix A of 10 CFR 50 (GDC 19).

PCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

PCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

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

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE from within the PCPB is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the PCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

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

LCO 3.4.14, "PCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in PCS LEAKAGE when

BASES

LCO

c. Identified LEAKAGE (continued)

the other is leaktight. If both valves leak and result in a loss of mass from the PCS, the loss must be included in the allowable identified LEAKAGE.

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 6). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

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

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

ACTIONS

A.1

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

BASES

ACTIONS (continued)

B.1 and B.2

If any pressure boundary LEAKAGE from within the PCPB exists or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying PCS LEAKAGE to be within the LCO limits ensures the integrity of the PCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an PCS water inventory balance.

The PCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. The Surveillance is modified by two Notes. Note 1 states that the SR is not required to be performed in MODES 3 and 4, until 12 hours of steady state operation have elapsed.

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

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "PCS Leakage Detection Instrumentation."

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

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

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

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 7. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 7).

BASES (continued)

REFERENCES

1. FSAR, Section 5.1.5
2. FSAR, Section 14.15
3. FSAR, Section 14.14
4. FSAR, Section 14.16
5. FSAR, Section 14.24
6. NEI 97-06, "Steam Generator Program Guidelines"
7. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

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

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

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

Specification 5.5.8, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.8, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.8. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

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

BASES (continued)

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "PCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via the Main Steam Safety Valves and Atmospheric Dump Valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.3 gpm or is assumed to increase to 0.3 gpm as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "PCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

LCO

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

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

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

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

BASES

LCO (continued)

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

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

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 0.3 gpm per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "PCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

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

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

ACTIONS

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

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is

BASES

ACTIONS

A.1 and A.2 (continued)

discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1 (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

BASES (continued)

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines"

2. 10 CFR 50 Appendix A, GDC 19

3. 10 CFR 100

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines"

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage and Supply

BASES

BACKGROUND

The Condensate Storage and Supply provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Primary Coolant System (PCS). The Condensate Storage Tank (CST) and the Primary Makeup Storage Tank (T-81) provide a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5, "Auxiliary Feedwater (AFW) System"). Three AFW pumps take a suction from a common line from the CST. T-81 provides makeup to the CST either by use of a pump or by gravity flow. Backup sources from the Service Water System (SWS) and Fire Water System an provide additional water supply to the AFW pump suctions if the normal source is lost. SWS provides an emergency source to AFW pump P-8C, and the Fire Water System provides an emergency source to AFW pumps P-8A and P-8B. The steam produced is released to the atmosphere by the Main Steam Safety Valves (MSSVs) or the atmospheric dump valves. The AFW pumps operate with a continuous recirculation to the CST.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the turbine bypass valve. This has the advantage of conserving condensate while minimizing releases to the environment.

Because the CST is a principal component in removing residual heat from the PCS, it is designed to withstand earthquakes. The tornado protected supply is provided by the SWS and Fire Water System. The CST is designed to Seismic Category I requirements to ensure availability of the feedwater supply.

A description of the Condensate Storage and Supply is found in the FSAR, Section 9.7 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The Condensate Storage and Supply provides condensate to remove decay heat and to cool down the plant following all events in the accident analysis, discussed in the FSAR, Chapters 5 and 14. For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs followed by a cooldown to Shutdown Cooling (SDC) entry conditions at the design cooldown rate.

The Condensate Storage and Supply satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

To satisfy accident analysis assumptions, the CST and T-81 must contain sufficient cooling water to remove decay heat for 8 hours following a reactor trip from 2580.6 MWth. This amount of time allows for cool down of the PCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this the CST and T-81 must retain sufficient water to ensure adequate net positive suction head for the AFW pumps, and makeup for steaming required to remove decay heat.

OPERABILITY of the Condensate Storage and Supply System is determined by maintaining the combined tank levels at or above the minimum required volume.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the Condensate Storage and Supply is required to be OPERABLE.

In MODES 5 and 6, the Condensate Storage and Supply is not required because the AFW System is not required.

BASES

ACTIONS

A.1 and A.2

If the condensate volume is not within the limit, the OPERABILITY of the backup water supplies must be verified by administrative means within 4 hours and once every 12 hours thereafter.

OPERABILITY of the backup feedwater supplies must include verification of the OPERABILITY of flow paths from the Fire Water System and SWS to the AFW pumps, and availability of the water in the backup supplies. The Condensate Storage and Supply volume must be returned to OPERABLE status within 7 days, as the backup supplies may be performing this function in addition to their normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the Fire Water System and SWS. Additionally, verifying the backup water supplies every 12 hours is adequate to ensure the backup water supplies continue to be available. The 7 day Completion Time is reasonable, based on OPERABLE backup water supplies being available, and the low probability of an event requiring the use of the water from the CST and T-81 occurring during this period.

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per . . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.7.6 A.1 must be initially performed within 4 hours without any SR 3.0.2 extension, subsequent performances at the "Once per 12 hours" interval may utilize the 25% SR 3.0.2 extension.

B.1 and B.2

If the condensate volume cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the combination of CST and T-81 contain the required useable volume of cooling water. (This volume ≥ 100,000 gallons.) The 12 hour Frequency is based on operating experience, and the need for operator awareness of plant evolutions that may affect the Condensate Storage and Supply inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal CST and T-81 level deviations.

REFERENCES

1. FSAR, Section 9.7

B 3.9 REFUELING OPERATIONS

B 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation - High Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Primary Coolant System (PCS) as required by the Palisade Nuclear Plant design, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the PCS by circulating primary coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the PCS via the PCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of primary coolant through the SDC heat exchanger(s). Mixing of the primary coolant is maintained by this continuous circulation of primary coolant through the SDC System.

APPLICABLE SAFETY ANALYSES

If the primary coolant temperature is not maintained below 200°F, boiling of the primary coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the primary coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical.

The loss of primary coolant and the reduction of boron concentration in the primary coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the SDC System is required to be in operation in MODE 6, with the refueling cavity water level greater than or equal to the 647 ft elevation, to prevent this challenge. The LCO allows the removal of an SDC train from operation for short durations under the condition that the boron concentration of the primary coolant is not reduced.

This conditional allowance does not result in a challenge to the fission product barrier.

SDC and Coolant Circulation - High Water Level satisfies Criterion 4 of 10 CFR 50.36(c)(2).

BASES

LCO

Only one SDC train is required for decay heat removal in MODE 6, with the refueling cavity water level greater than or equal to the 647 ft elevation. Only one SDC train is required because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one SDC train must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC train consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the PCS temperature. The two SDC heat exchangers operate as a single unit. A separate OPERABLE SDC heat exchanger is required for each OPERABLE SDC train. The flow path starts in the Loop 2 PCS hot leg and is returned to at least one PCS cold leg.

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

The LCO is modified by two Notes. Note 1 allows the required operating SDC train to not be in operation for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the PCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and PCS to SDC isolation valve testing.

During this 1 hour period, decay heat is removed by natural circulation to the large mass of water in the refueling cavity. Note 2 allows the required SDC train to be made inoperable for ≤ 2 hours per 8 hour period for testing and maintenance provided one SDC train in operation providing flow through the reactor core, and the core outlet temperature is ≤ 200°F. The purpose of this Note is to allow the heat flow path from the SDC heat

BASES

exchanger to be temporarily interrupted for maintenance or testing on the Component Cooling Water or Service Water Systems.

LCO (continued)

During this 2 hour period, the core outlet temperature must be maintained ≤ 200°F. Requiring one SDC train to be in operation ensures adequate mixing of the borated coolant.

APPLICABILITY

One SDC train must be OPERABLE and in operation in MODE 6, with the refueling cavity water level greater than or equal to 647 ft elevation, to provide decay heat removal. The 647 ft elevation was selected because it corresponds to the elevation requirement established for fuel movement in LCO 3.9.6, "Refueling Cavity Water Level." Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, "Primary Coolant System (PCS)." SDC train requirements in MODE 6, with the refueling cavity water level less than the 647 ft elevation are located in LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

ACTIONS

SDC train requirements are met by having one SDC train OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If one required SDC train is inoperable or not in operation, actions shall be immediately initiated and continued until the SDC train is restored to OPERABLE status and to operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

A.2

If SDC train requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur through the addition of water with a lower boron concentration than that contained in the PCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

BASES

ACTIONS (continued)

A.3

If SDC train requirements are not met, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural circulation to the heat sink provided by the water above the core. A minimum refueling cavity water level equivalent to the 647 ft elevation provides an adequate available heat sink. Suspending any operation that would increase the decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.4

If SDC train requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed to prevent fission products, if released by a loss of decay heat removal event, from escaping to the environment. The 4 hour Completion Time is based on the low probability of the coolant boiling in that time and allows time for fixing most SDC problems.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the SDC train is in operation and circulating primary coolant. The flow rate is sufficient to provide decay heat removal capability and to prevent thermal and boron stratification in the core. The 1000 gpm flow rate has been determined by operating experience rather than analysis. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the SDC System.

REFERENCES

1. FSAR, Sections 6.1 and 14.3

B 3.9 REFUELING OPERATIONS

B 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Primary Coolant System (PCS), as required by the Palisades Nuclear Plant design, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the PCS by circulating primary coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the PCS via the PCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of primary coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the primary coolant is maintained by this continuous circulation of primary coolant through the SDC System.

APPLICABLE SAFETY ANALYSES

If the primary coolant temperature is not maintained below 200°F, boiling of the primary coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the primary coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical.

The loss of primary coolant and the reduction of boron concentration in the primary coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the SDC System are required to be OPERABLE, and one train is required to be in operation in MODE 6, with the refueling cavity water level less than the 647 ft elevation to prevent this challenge.

SDC and Coolant Circulation - Low Water Level satisfies Criterion 4 of 10 CFR 50.36(c)(2).

BASES

LCO

In MODE 6, with the refueling cavity water level less than the 647 ft elevation, both SDC trains must be OPERABLE. Additionally, one train of the SDC System must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of primary coolant temperature.

An OPERABLE SDC train consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the PCS temperature. The two SDC heat exchangers operate as a single unit. A separate OPERABLE SDC heat exchanger is required for each OPERABLE SDC train. The flow path starts in one of the PCS hot legs and is returned to the PCS cold legs.

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

Both SDC pumps may be aligned to the safety injection refueling water tank to support filling the refueling cavity or for performance of required testing.

APPLICABILITY

Two SDC trains are required to be OPERABLE, and one SDC train must be in operation in MODE 6, with the refueling cavity water level less than the 647 ft elevation to provide decay heat removal. Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, "Primary Coolant System." MODE 6 requirements, with the refueling cavity water level greater than or equal to the 647 ft elevation are covered in LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level."

BASES

ACTIONS

A.1 and A.2

If one SDC train is inoperable, action shall be immediately initiated and continued until the SDC train is restored to OPERABLE status, or until a water level of greater than or equal to the 647 ft elevation is established. When the water level is established at the 647 ft elevation or greater, the plant conditions will change so that LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level," is applicable, and only one SDC train is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no SDC train is in operation or no SDC trains are OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the PCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

B.2

If no SDC train is in operation or no SDC trains are OPERABLE, action shall be initiated immediately and continued without interruption to restore one SDC train to OPERABLE status and operation. Since the plant is in Conditions A and B concurrently, the restoration of two OPERABLE SDC trains and one operating SDC train should be accomplished expeditiously.

B.3

If no SDC train is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed Immediately. With the SDC train requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one SDC train is operating and circulating primary coolant. The flow rate is sufficient to provide decay heat removal capability and to prevent thermal and boron stratification in the core.

In addition, during operation of the SDC train with the water level in the vicinity of the reactor vessel nozzles, the SDC train flow rate determination must also consider the SDC pump suction requirements. The 1000 gpm flow rate has been determined by operating experience rather than analysis. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the SDC System in the control room.

SR 3.9.5.2

Verification that the required pump is OPERABLE ensures that an additional SDC pump can be placed in operation, if needed, to maintain decay heat removal and primary coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. FSAR, Sections 6.1 and 14.3