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HDI PNP 2023-030

10 CFR 50.90

December 14, 2023

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

Palisades Nuclear Plant
NRC Docket No. 50-255
Renewed Facility Operating License No. DPR-20

Subject: License Amendment Request to Revise Renewed Facility Operating License and Permanently Defueled Technical Specifications to Support Resumption of Power Operations

In accordance with Title 10 of the Code of Federal Regulations, Part 50, Section 90 (10 CFR 50.90), *Application for amendment of license, construction permit, or early site permit*, Holtec Decommissioning International, LLC (HDI) on behalf of Holtec Palisades LLC, hereby requests U. S. Nuclear Regulatory Commission (NRC) review and approval of a proposed license amendment request (LAR) to revise the Palisades Nuclear Plant (PNP) Renewed Facility Operating License (RFOL) DPR-20. The proposed LAR would revise the RFOL, Appendix A, Permanently Defueled Technical Specifications (PDTS), and Appendix B Environmental Protection Plan (EPP) to reflect the resumption of power operations at PNP.

In Reference 1, Entergy Nuclear Operations, Inc. notified the NRC that it had permanently ceased operations and permanently removed fuel from the reactor vessel at PNP. Upon docketing the 10 CFR 50.82, *Termination of license*, paragraph a, subparagraph 1, 10 CFR 50.82(a)(1) certifications 10 CFR 50.82(a)(2) no longer authorizes operation of the reactor, or emplacement or retention of fuel into the reactor vessel. However, shortly after PNP transitioned to a decommissioning facility, Holtec Palisades LLC assumed ownership of PNP (Reference 2) and given the support from the Governor of the State of Michigan, HDI commenced a project to return PNP to a power operations plant. The regulatory part of this project as described in an HDI letter dated March 13, 2023 (Reference 3), has identified the regulatory path to reinstate the power operations licensing basis (POLB) to resume power operations through a series of licensing submittals referred to as a “regulatory framework.” HDI intends to submit additional licensing actions over the next several months to reinstate the Plant POLB. In this submittal HDI proposes to restore the PNP power operations technical specifications (POTS) which is a part of this regulatory framework.

The PNP repower regulatory framework consists of this LAR, a LAR to revise PDTS administrative requirements, an emergency plan LAR, an exemption to 10 CFR 50.82(a)(2) (Reference 4), and a license transfer order for PNP operating authority (Reference 5). As discussed in Reference 4, to coordinate implementing this requested amendment, after receipt of NRC approvals needed to return PNP to power operations, HDI is proposing to submit a notification of transition to power operations to the NRC that will docket HDI's satisfaction of the implementation conditions for license transfer, 10 CFR 50.82(a)(2) exemption, and license amendments. Upon docketing this notification letter, PNP will transition from a facility in decommissioning back to a power operations plant.

HDI is currently targeting the implementation of the POTS in the third quarter of 2025. To support this schedule, HDI respectfully requests that the NRC review the enclosed LAR on a schedule that that will permit approval of the proposed LAR by January 31, 2025, and that the proposed amendment become effective upon docketing the transition notification letter, with a 30-day implementation period.

The proposed changes to the PNP PDTS are in accordance with 10 CFR 50.36, *Technical specifications*, 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes to the PNP EPP are in accordance with 10 CFR 50.36b, *Environmental conditions*, paragraph b, 10 CFR 50.36b(b).

The enclosure to this letter provides a detailed description and evaluation of the proposed changes for PNP. Attachment 1 to the enclosure contains a mark-up of the current RFOL, PDTS, and EPP pages. Attachment 2 to the enclosure contains the retyped RFOL, TS, and EPP containing the proposed changes. Attachment 3 to the enclosure contains the proposed changes to the TS Bases. The proposed TS Bases changes are provided for information and will be incorporated in accordance with the TS Bases Control Program upon implementation of the approved amendment.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a), *Notice for public comment*, subparagraph (1), using the standards in 10 CFR 50.92, *Issuance of amendment*, paragraph (c), and it has been determined that the changes involve no significant hazards consideration. The basis for this determination is included in the enclosure.

In accordance with 10 CFR 50.91(b), *State consultation*, HDI is notifying the State of Michigan of this proposed LAR by transmitting a copy of this letter, with its enclosure, to the designated State of Michigan official.

If you have any questions regarding this submittal, please contact Jim Miksa, regulatory assurance engineer, at (269) 764-2945.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 14, 2023.

Respectfully,

Jean A. Fleming Digitally signed by Jean A. Fleming
Date: 2023.12.14 09:26:30 -05'00'

Jean A. Fleming
Vice President, of Licensing, Regulatory Affairs & PSA
Holtec International

References:

1. Entergy Nuclear Operations, Inc. letter to U.S. Nuclear Regulatory Commission, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel," dated June 13, 2022 (ADAMS Accession No. ML22164A067)

2. U.S. Nuclear Regulatory Commission letter to Holtec International, “Palisades Nuclear Plant and Big Rock Point Plant – Issuance of Amendment Nos. 129 and 273 re: Order Approving Transfer of Licenses and Conforming Administrative License Amendments,” dated June 28, 2022 (ADAMS Accession No. ML22173A173)

3. Holtec Decommissioning International, LLC letter to U.S. Nuclear Regulatory Commission, "Regulatory Path to Reauthorize Power Operations at the Palisades Nuclear Plant" dated March 13, 2023 (ADAMS Accession No. ML23072A404)

4. Holtec Decommissioning International, LLC letter to U.S. Nuclear Regulatory Commission, "Request for Exemption from Certain Termination of License Requirements if 10 CFR 50.82" dated September 28, 2023 (ADAMS Accession No. ML23271A140)

5. Holtec Decommissioning International, LLC letter to U.S. Nuclear Regulatory Commission, "Application for Order Consenting to Transfer of Control of License and Approving Conforming License Amendments" dated December 6, 2023 (ADAMS Accession Nos. ML23340A161, ML23340A162)

Enclosure: Evaluation of the Proposed Changes

Enclosure Attachments:

1. Proposed Changes (mark-up) to Palisades Plant Renewed Facility Operating License DPR-20, Appendix A Permanently Defueled Technical Specifications, and Appendix B Environmental Protection Plan Pages

2. Page Change Instructions and Retyped Pages for the Palisades Plant Renewed Facility Operating License DPR-20, Appendix A Technical Specifications, and Appendix B Environmental Protection Plan

3. Proposed Technical Specifications Bases Changes (for information only)

cc: NRC Region III Regional Administrator
NRC Decommissioning Inspector – PNP
NRC Project Manager PNP
Designated Michigan State Official

Enclosure

HDI PNP 2023-030

Evaluation of the Proposed Changes

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EVALUATION OF THE PROPOSED CHANGES

1.0 SUMMARY DESCRIPTION

In accordance with Title 10 of the Code of Federal Regulations, Part 50, Section 90 (10 CFR 50.90), *Application for amendment of license, construction permit, or early site permit*, Holtec Decommissioning International, LLC (HDI) on behalf of Holtec Palisades, LLC hereby requests U. S. Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) to revise the Palisades Nuclear Plant (PNP) Renewed Facility Operating License (RFOL) DPR-20. The proposed license amendment would revise the RFOL, Appendix A, Permanently Defueled Technical Specifications (PDTS), and Appendix B Environmental Protection Plan (EPP). The proposed changes reinstate PNP license requirements that were removed based on docketing the 10 CFR 50.82(a)(1) certifications of permanent cessation of power operations and permanent removal of fuel from the reactor vessel to support a return of the PNP plant to power operations. The requested changes involve no significant hazards consideration. HDI requests approval of the proposed LAR by January 31, 2025, and that the proposed amendment become effective upon docketing the transition notification letter, with a 30-day implementation period.

2.0 DETAILED DESCRIPTION

2.1 Reason for Proposed Change

In Reference 1, Entergy Nuclear Operations, Inc. notified the NRC that it decided to permanently cease operations at PNP no later than May 31, 2022. Certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel were submitted to the NRC in accordance with 10 CFR 50.82(a)(1)(i) and (ii), respectively, and were docketed (Reference 2). Upon docketing the 10 CFR 50.82(a)(1) certifications 10 CFR 50.82(a)(2) no longer authorizes operation of the reactor, or emplacement or retention of fuel into the reactor vessel. The regulatory framework for the reauthorization of power operations at PNP includes submitting a request for exemption from 10 CFR 50.82(a)(2) to remove the restriction that prohibits operation of the PNP reactor, or emplacement or retention of fuel into the PNP reactor vessel (Reference 3). This restriction, imposed by the voluntary docketing of the 10 CFR 50.82(a)(1) certifications, was used as the basis for licensing actions that allowed relaxation of power operation license requirements at PNP. Implementation of the NRC approved licensing actions included revising the PNP licensing basis to accurately reflect the status and reduced risk of a facility in decommissioning. No major decommissioning activities occurred at PNP to support this transition, and none have occurred since. There are no physical changes to facility design proposed or required to support this exemption. LAR along with the referenced exemption, an operating authority transfer order, and LARs to the RFOL administrative requirements and Emergency Plan are required to support reinstatement of the PNP power operations licensing basis that was in effect just prior to the 10 CFR 50.82(a)(1) certifications.

As described above, this LAR is necessary to reinstate the PNP RFOL, TS, and the EPP that were in effect just prior to the 10 CFR 50.82(a)(1) certifications to support returning PNP to a power operations licensing basis (POLB). To retain a clear connection between the RFOL decommissioning license amendments (References 4 and 5) and the license amendments to return PNP to power operations, HDI has elected to submit two separate LARs to revise the RFOL, PDTS and EPP. One is this LAR, which reinstates the TS needed for resumption of power operation. A second LAR reinstates TS for certain administrative requirements for plant staff. Although both LARs must be approved and implemented prior to the resumption of power operation, they are not linked.

2.2 Description of Proposed Change

This LAR proposes to revise the PNP RFOL, the PDTS, and the EPP. The proposed changes are consistent with the previously approved PNP power operations technical specifications (POTS) that allowed emplacement of fuel into the reactor vessel and power operations at PNP. The proposed changes would revise certain requirements contained within the PNP RFOL and PDTS to reinstate requirements that are necessary for power operation and revise or remove requirements that would no longer be applicable. The proposed EPP changes are editorial/administrative to more accurately reflect a power operations plant. The proposed changes to the PNP PDTS are in accordance with 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes to the PNP EPP are in accordance with 10 CFR 50.36b(b).

The Updated Final Safety Analysis Report (UFSAR), now titled the Defueled Safety Analysis Report (DSAR), will be updated, via the 10 CFR 50.59, *Changes, tests and experiments*, process to reflect the docketed version that was in effective prior to the 10 CFR 50.82(a)(1) certifications, PNP UFSAR Revision 35 (Reference 6). Any DSAR retained changes to UFSAR Revision 35 have been or will be evaluated via the 50.59 process against UFSAR Revision 35 to determine if NRC approval is required prior to exiting the period of decommissioning. This will include reinstatement of accident analyses and the safety reclassification of systems, structures, and components (SSCs), required to support the PNP power operations licensing basis (POLB). Changes made to the UFSAR after Revision 35 will be evaluated for retention, to the extent appropriate for an operating plant. The DSAR change back to the PNP POLB UFSAR will be accomplished under the 10 CFR 50.59 process and be implemented coincident with the associated license amendments.

HDI has submitted an application to the NRC for the transfer of the PNP operating authority from HDI to a new entity (Reference 11). Ownership of the PNP license will remain with Holtec Palisades LLC. This application is necessary because the order that transferred operating authority to HDI limits HDI to the performance of spent fuel management and decommissioning activities at PNP (Reference 7). The license transfer request changes are independent of the changes proposed in this enclosure and, as such, the changes proposed by the license transfer request are not included in the RFOL and PDTS markups of this LAR.

3.0 TECHNICAL EVALUATION

This LAR proposes modifications to the PNP RFOL, the PDTS, and the EPP to support reinstatement of power operation at PNP.

The regulatory requirements related to the content of TS are promulgated in 10 CFR 50.36, *Technical Specifications*. As detailed in a subsequent section of this LAR, this regulation lists the criteria to define the scope of items that must be included in TS. The scope of systems structures and components (SSCs) and parameters that must have Limiting Conditions for Operation (LCO) included in the PNP TS to support the PNP POLB are those needed to address the reinstated UFSAR Revision 35 postulated design basis accidents (DBAs), so that the consequences of the DBAs are maintained within acceptable limits. PNP TS LCOs that were removed in Amendment 272 (Reference 8) will be reinstated to support the PNP POLB, UFSAR Revision 35 (Reference 6).

3.1 Accident and Transient Analyses Applicable to the Proposed Change

As stated in Reference 8, Chapter 14 of the PNP UFSAR (Reference 6) and license Amendment No. 226 (Reference 9) describe the postulated DBA and transient scenarios applicable to PNP during power operations and when fuel is in the reactor vessel. These scenario's analyses demonstrate that the PNP plant design supports safe power operations and retention of fuel in the reactor vessel, and that radiological consequences from the postulated accident scenarios do not exceed the regulatory requirements of 10 CFR 50.67 or 10 CFR Part 100, as applicable. Two basic groups of events are pertinent to safety: abnormal operational transients and postulated DBAs. The analyses of the abnormal operational transients evaluate the ability of the plant protection features to ensure that during these transients no fuel damage occurs, and the primary coolant system (PCS) pressure limit is not exceeded. The safety design limits require that damage to the fuel be limited and that no nuclear system process barrier damage results from any abnormal operational occurrence. Thus, analysis of this group of events evaluates the features that protect the first two radioactive material barriers. The radioactive material barriers are the fuel cladding, primary coolant system and the Containment. Analyses of the events in the second group, postulated DBAs, evaluate situations that require functioning of the engineered safeguards systems in order to protect the fission product barriers, including containment, in order to mitigate the offsite radiological consequences.

PNP UFSAR Revision 35, Chapter 14 (Reference 6) contains the DBA and transient scenarios applicable to PNP during plant operations. It also describes the analyses that were performed to demonstrate that the plant could be operated safely and that radiological consequences from postulated DBAs do not exceed the applicable limits. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products to the PCS. Many of the accident scenarios postulated in the UFSAR involve failures or malfunctions of systems which could affect the reactor core. Once the reactor is permitted to be refueled and resume power operation, these UFSAR events are applicable.

The analyses in UFSAR Chapter 14 provide results that are compared to regulatory acceptance criteria and are integral to the plant's design and licensing basis. These analyses demonstrate the integrity of the fission product barriers, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of DBAs and transients. Certain systems, structures, and components (SSCs) are credited in these analyses for the purpose of mitigating the DBAs or transients. These SSCs are considered for inclusion in TS per the requirements of 10 CFR 50.36(c).

As noted above, the UFSAR will be reinstated to Revision 35 (Reference 6) which is the last docketed version of the UFSAR that was in effect prior to the 10 CFR 50.82(a)(1) certifications. This includes reinstatement of the DBA and transient scenarios in Chapter 14. These scenarios and analyses are reinstated as they existed in Revision 35 of the UFSAR. No changes to these analyses are made as part of the return to power for PNP. As a normal part of evaluating core designs and operational parameters, reviews of Chapter 14 sections are performed for each fuel reload cycle by the nuclear fuel vendor to determine if the proposed core design is still bounding or if it needs to be reanalyzed (disposition of events). The UFSAR is updated accordingly, as needed. Therefore, future changes to Chapter 14 of the UFSAR may occur separately from the licensing actions necessary to resume power operation and will be evaluated under the 10 CFR 50.59 process against the PNP POLB.

In the PDTS LAR (Reference 4), PNP discussed four accident analyses that remained valid for a permanently shut down and defueled condition. They are the postulated cask drop accident, fuel handling incident in the fuel handling building, the liquid waste incident, and the waste gas incident. These four accident analyses will be retained and revised as necessary to the UFSAR Revision 35 versions that were in place prior to the PDTS amendment 272 (Reference 8).

In the PDTS LAR (Reference 4) the postulated cask drop accident was not reanalyzed for the permanently defueled condition. In lieu of revising the cask drop accident analysis, restrictions were placed on the movement of a fuel cask (License Condition 2.C.(5)) to support removal of TS 3.7.10, *Control Room Ventilation (CRV) Filtration*, TS 3.7.11, *Control Room Ventilation (CRV) Cooling*, and TS 3.7.12, *Fuel Handling Area Ventilation System*, thereby ensuring continued compliance with the radiological consequences described for a postulated cask drop during shutdown and defueled conditions. Therefore, since the postulated cask drop analysis described in UFSAR Revision 35 was not revised to support the PDTS LAR, it is proposed to remain unchanged as the analysis of record with the proposed deletion of License Condition 2.C.(5) supported by the reinstatement of TS 3.7.10, TS 3.7.11, and TS 3.7.12.

The fuel handling accident (FHA) was updated to add a sensitivity analysis to determine the fuel decay time at the time of a FHA (determined to be 17 days) that is required to no longer credit isolation of the Control Room (CR) ventilation system and radionuclide removal through the CR and SFP ventilation systems. This sensitivity analysis addressed only a fuel assembly drop in the spent fuel pool and did not revise the previously approved fuel handling accident in containment as described in UFSAR Revision 35. This previous version of the FHA as discussed in UFSAR Revision 35 will be reinstated as the analysis of record and includes a fuel assembly drop in containment as well as in the spent fuel pool. Additionally, with the reinstatement of TS 3.7.10, TS 3.7.11, and TS 3.7.12 the sensitivity analysis for fuel decay time is no longer necessary.

The accidental release of liquid waste was not reanalyzed for the permanently defueled condition. Accidental discharge of radioactive liquid from a volume control tank (VCT) rupture, primary makeup storage tank (T-90) failure, or utility water storage tank (T-91) failure to the circulating water canal will continue to be controlled by administrative process, system design, and system monitoring during normal operation. The same controls will remain in place in the operating condition as were previously credited both during operation and the permanently defueled condition. Therefore, no change is proposed to the accidental release of liquid waste incident, and it will reinstated as described in UFSAR Revision 35.

The analysis for accidental release of waste gas consists of two parts, waste gas decay tank failure and rupture of the volume control tank. For the permanently defueled condition the waste gas decay tank failure analysis was revised to support deletion of TS 3.7.10 and TS 3.7.11 by demonstrating that the dose consequences are bounded by the FHA. This LAR proposes to reinstate TS 3.7.10 and TS 3.7.11 to support the approved waste gas decay tank rupture analysis described in UFSAR Revision 35. The volume control tank rupture analysis that was no longer applicable in the permanently defueled condition will again be applicable in the power operations condition and will be reinstated without revision to the version described in UFSAR Revision 35. The reinstated analysis, without credit for release path filtration, demonstrates EAB, LPZ, and CR dose criteria are not exceeded. Therefore, reinstatement of the UFSAR Revision 35 waste gas incident supports power operations at PNP.

A list of the PNP UFSAR Chapter 14 safety analysis DBAs and transients is provided in Table 3-1. Analyses are reviewed on a cycle specific basis as appropriate and are addressed by the fuel vendor as described in UFSAR Section 14.1. Analyses of radiological impacts for certain events were reviewed and approved by the NRC in Amendment 226 (Reference 9). This amendment was the full implementation of an alternate source term in accordance with the guidance in Regulatory Guide 1.183 (Reference 10). The alternate source term amendment approved by the NRC will remain the analysis of record and is unchanged since approved by amendment 226. No changes were made to these analyses during the transition to decommissioning.

Table 3-1 - PNP DBAs and Transients

UFSAR Section	DBA or Transient
14.2	Uncontrolled Control Rod Withdrawal
14.3	Boron Dilution
14.4	Control Rod Drop
14.5	Core Barrel Failure
14.6	Control Rod Mis-operation
14.7	Decreased Reactor Coolant Flow
14.8	Start-up of an Inactive (Primary Coolant Pump) Loop
14.9	Excessive Feedwater Incident
14.10	Increase in Steam Flow (Excess Load)
14.11	Postulated Cask Drop Accidents
14.12	Loss of External Load
14.13	Loss of Normal Feedwater
14.14	Steam Line Rupture Incident
14.15	Steam Generator Tube Rupture with a Loss of Offsite Power
14.16	Control Rod Ejection
14.17	Loss of Coolant Accident
14.18	Containment Pressure and Temperature Analysis
14.19	Fuel Handling Incident
14.20	Liquid Waste Incident
14.21	Waste Gas Incident
14.22	Maximum Hypothetical Accident
14.23	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment
14.24	Control Room Radiological Habitability

These DBAs and incidents are once again applicable to PNP in a power operating condition. They are the same ones previously described in UFSAR, Revision 35. When reinstated, these analyses will reflect the licensing basis as it existed just prior to the 10 CFR 50.82(a)(1) certifications to support returning PNP to a POLB. These accident analyses form the basis for the TS that existed prior to the defueled period, and they now form the basis for the TS proposed in this LAR.

3.2 Evaluation of the Proposed Change

HDI proposes to modify the PNP RFOL, PDTS, and EPP as shown below. Each section that is proposed to be changed is identified, the proposed changes are shown, and the basis for each change is given. Changes to the RFOL are listed first followed by changes to the PDTS and EPP. If appropriate, proposed deletions are shown with strikethrough and additions are shown in bold italics. TS sections that were deleted in their entirety with issuance of Amendment 272, PDTS, (Reference 8) are proposed for reinstatement without change to create TS sections consistent with power operation. The addition of these complete TS sections is described below.

Attachment 1 to this enclosure contains a mark-up of the current RFOL, PDTS, and EPP pages. The proposed changes to the PDTS are considered a major rewrite with the addition of numerous TS. TS sections that are added in their entirety are listed by number and title, however the inserted TS pages are not included with the marked-up pages in Attachment 1. In addition, the following editorial changes are not shown in the marked-up RFOL and PDTS in Attachment 1 because they do not affect the technical content of the RFOL or the PDTS:

- Reformatting (margins, font, tabs, line spacing, etc.) content to create a continuous electronic file,
- Renumbering of pages, where appropriate, and
- Removal of historical amendment numbers.

Attachment 2 of this enclosure provides the re-typed and added pages to reflect the proposed changes in their entirety.

The markups of the TS Bases, including TS Bases sections that will be added in their entirety are provided in Attachment 3. They are provided for information only. Upon approval of this amendment, changes to the TS Bases will be incorporated in accordance with PNP TS 5.5.12, "Technical Specifications (TS) Bases Control Program."

3.2.1 Proposed Changes to the PNP Renewed Facility Operating License

Changes proposed to the RFOL are necessary to reinstate the license conditions that were removed by license amendment 272 (Reference 8) due to docketing the 10 CFR 50.82 decommissioning certifications as conditioned by the exemption to 10 CFR 50.82(a)(2) (Reference 3). A License Condition removed in Amendment 272 because it was identified as superseded (i.e., original License Condition 1.B) will not be reinstated. License Conditions removed in Amendment 272 because they were identified as historical Conditions will not be reinstated (i.e., original License Conditions 2.C(4), 2.C(7), 2.H, and 2.I). See Reference 4 for the discussion and approval of the superseded and historical designations. No changes are proposed to the previously approved reinstated license conditions.

As previously discussed, HDI has submitted an application to the NRC for the transfer of the PNP operating authority from HDI to a new entity (Reference 11). Ownership of the PNP license will remain with Holtec Palisades LLC. Note, references to "HDI" are replaced by bracketed Palisades Energy, LLC, or Palisades Energy (e.g. [Palisades Energy]) to reflect the change in operating authority per license transfer application conforming amendments (Reference 11). They will be changed coincident with the implementation of the operating authority transfer through issuance of conforming amendments. They are not addressed in this submittal.

License Condition 2.B.(1)	
Current License Condition 2.B.(1)	Proposed License Condition 2.B.(1)
Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", (a) Holtec Palisades to possess and use, and (b) HDI to possess and use the facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;	Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", (a) Holtec Palisades to possess and use, and (b) ~~HDI~~ ***[Palisades Energy]*** to possess~~, and~~ use ***and operate,*** the facility ***as a utilization facility*** at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;
Basis This License Condition is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this license condition is reinstated to acknowledge that the plant will resume power operation as a utilization facility defined by 10 CFR Part 50.	
License Condition 2.B.(2)	
Current License Condition 2.B.{2}	Proposed License Condition 2.B.{2}
HDI, pursuant to the Act and 10 CFR Parts 40 and 70, to possess source and special nuclear material that was used as reactor fuel, in accordance with the limitations for storage as described in the Updated Final Safety Analysis Report, as supplemented and amended;	~~HDI~~ ***[Palisades Energy]***, pursuant to the Act and 10 CFR Parts 40 and 70, to ***receive,*** possess***, and use*** source and special nuclear material ~~that was used~~ as reactor fuel, in accordance with the limitations for storage ***and amounts required for reactor operation,*** as described in the Updated Final Safety Analysis Report, as supplemented and amended;
Basis This License Condition is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this license condition is reinstated to permit authorization for receipt and use of special nuclear material (SNM) as reactor fuel, add the reference to use of the SNM for reactor operations, and allow possession of SNM as reactor fuel in the amount required for reactor operation. With the resumption of power operation, PNP has a need to receive SNM in the form of reactor fuel and use SNM as reactor fuel for reactor operations.	

License Condition 2.B.(3)	
Current License Condition 2.B.(3)	Proposed License Condition 2.B.(3)
HDI pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources that were used for reactor startup, sealed sources that were used for reactor instrumentation and are used in the calibration of radiation monitoring equipment, and that were used as fission detectors in amounts as required;	~~HDI~~ ***[Palisades Energy]*** pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources ~~that were used~~ for reactor startup, ~~sealed sources that were used for~~ reactor instrumentation***,*** ~~and are used in the calibration of~~ radiation monitoring equipment ***calibration***, and ~~that were used as~~ fission detectors in amounts as required;
Basis	
This License Condition is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this license condition is reinstated to authorize the receipt and use of byproduct, source, and SNM as sealed neutron sources for reactor startup, reactor instrumentation, and fission detectors. This reinstatement is consistent with the fact that PNP will become a power operation plant and use of this byproduct material is necessary for plant functions as described.	

License Condition 2.B.(5)	
Current License Condition 2.B.(5)	Proposed License Condition 2.B.(5)
HDI pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials that were produced by the operations of the facility.	~~HDI~~ ***[Palisades Energy]*** pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials ***as may be*** ~~that were~~ produced by the operations of the facility.
Basis	
This License Condition is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this license condition is reinstated to reflect that PNP will become a power operation plant and produce byproduct and special nuclear materials in the course of plant operation.	

License Condition 2.C.(1)	
Current License Condition 2.C.(1)	Proposed License Condition 2.C.(1)
[deleted]	~~[deleted]~~ ***[Palisades Energy] is authorized to operate the facility at steady state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.***
Basis	
This License Condition is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this License Condition is reinstated in its entirety to reflect the operating condition and technical specifications for an operating plant. The maximum power level is defined in this License Condition and is consistent with that used to support the accident analyses evaluated in the UFSAR.	

License Condition 2.C.(2)	
Current License Condition 2.C.(2)	Proposed License Condition 2.C.(2)
The Technical Specifications contained in Appendix A, as revised through Amendment No. 273, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. HDI shall maintain the facility in accordance with the Technical Specifications and the Environmental Protection Plan.	The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~273~~**XXX**, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. ~~HDI~~ ***[Palisades Energy]*** shall ~~maintain~~ ***operate*** the facility in accordance with the Technical Specifications and the Environmental Protection Plan.
Basis	
This License Condition is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this License Condition is reinstated to reflect that PNP will become a power operation plant. The current amendment number, 273, is deleted and replaced with "XXX" which acts as a placeholder for the new amendment number associated with the approval of this LAR.	

<table>
<tr><th colspan="2">License Condition 2.C.(3)</th></tr>
<tr><td><u>Current License Condition 2.C.(3)</u>
[deleted]</td><td><u>Proposed License Condition 2.C.(3)</u>
~~[deleted]~~
<u>Fire Protection</u>
[Palisades Energy] shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment requests ….[for entirety of text proposed for addition, see Attachment 1]… 2. The licensee shall implement the modifications to its facility, as described in Table S-2, “Plant Modifications Committed,” of Entergy Nuclear Operations, Inc. (ENO) letter PNP 2019-028 dated May 28, 2019, to complete the transition to full compliance with 10 CFR 50.48(c) before the end of the refueling outage following the fourth full operating cycle after NRC approval. ….[for entirety of text proposed for addition, see Attachment 1]… will be completed once the related modifications are installed and validated in the PRA model.</td></tr>
<tr><td colspan="2">Basis
This License Condition is proposed for reinstatement in its entirety, with editorial revision, to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this License Condition is reinstated to provide requirements for implementation of a fire protection program that complies with 10 CFR 50.48(a) and 10 CFR 50.48(c), including requirements for risk informed changes that may be made without prior NRC approval, other changes that may be made without prior NRC approval, and transition License Conditions. See Attachment 1 for the complete License Condition. The revision to the earlier version of the License Condition is to spell out the first occurrence of ENO as Entergy Nuclear Operations, Inc. (ENO) when referring to letters previously issued.
10 CFR 50.48(a) and 10 CFR 50.48(c) apply to holders of operating licenses issued under Part 50. The conditions specified in 2.C.(3) include consideration of risk metrics for core damage frequency and large early release frequency, which are associated with power operation. This License Condition, which is based on maintaining an operational fire protection program in accordance with 10 CFR 50.48, with the ability to achieve and maintain safe shut down of the reactor in the event of a fire, will become applicable at PNP with the resumption of power operation. Therefore, License Condition 2.C.(3) is reinstated to reflect power operation of the facility.</td></tr>
</table>

<table>
<tr><th colspan="2">License Condition 2.C.(5)</th></tr>
<tr><td><u>Current License Condition 2.C.(5)</u>
Movement of a fuel cask in or over the spent fuel pool is prohibited when irradiated fuel assemblies decayed less than 90 days are in the spent fuel pool.</td><td><u>Proposed License Condition 2.C.(5)</u>
<s>Movement of a fuel cask in or over the spent fuel pool is prohibited when irradiated fuel assemblies decayed less than 90 days are in the spent fuel pool.</s>
[deleted]</td></tr>
<tr><td colspan="2">Basis
This License Condition was added to prohibit movement of a fuel cask in or over the spent fuel pool when irradiated fuel assemblies with less than 90 days decay time are in the spent fuel pool. Previously, TS 3.7.10, "Control Room Ventilation (CRV) Filtration, " and TS 3.7.11, "Control Room Ventilation (CRV) Cooling," were required during movement of a fuel cask in or over the SFP; TS 3.7.12, "Fuel Handling Area Ventilation System," was required during movement of a fuel cask in or over the SFP when fuel assemblies with less than 90 days decay time are in the fuel handling building. These TS were deleted in Amendment 272; therefore, this License Condition was needed to control the timing of cask movement in or over the spent fuel pool.

TS 3.7.10, TS 3.7.11, and TS 3.7.12 are being reinstated in the TS as described below. They will contain the limits on cask movement in or over the spent fuel pool. This License Condition is no longer needed to control cask movement. Reinstatement of these TS will ensure an appropriate spent fuel assembly decay time requirement is maintained and the associated analyses presented in UFSAR, Revision 35, Table 14.1-6 remains bounding.</td></tr>
<tr><th colspan="2">License Condition 2.C.(8)</th></tr>
<tr><td><u>Current License Condition 2.C.(8)</u>
[deleted]</td><td><u>Proposed License Condition 2.C.(8)</u>
<s>[deleted]</s>
Amendment 257 authorizes the implementation of 10 CFR 50.61a in lieu of 10 CFR 50.61.</td></tr>
<tr><td colspan="2">Basis
This License Condition is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this License Condition is reinstated in its entirety to reflect the regulatory requirements for an operating plant. 10 CFR 50.61a, Fracture toughness requirements for protection against pressurized thermal shock events, applies to pressurized water reactors for which an operating license has been issued.</td></tr>
</table>

<table>
<tr><th colspan="2">License Condition 2.D</th></tr>
<tr><td><u>Current License Condition 2.D</u>
[deleted]</td><td><u>Proposed License Condition 2.D</u>
~~[deleted]~~
The facility has been granted certain exemptions from Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors." This section contains leakage test requirements, scheduled and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted in a letter dated December 6, 1989.
These exemptions granted pursuant to 10 CFR 50.12, are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.</td></tr>
<tr><td colspan="2">Basis
This License Condition is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this License Condition is reinstated in its entirety to reflect the operating condition of the facility. During power operation, the containment plays a role in mitigating the consequences of the DBAs discussed in the UFSAR, Revision 35.
10 CFR Part 50, Appendix J provides leakage test requirements for the containment. The exemptions previously approved by the NRC for use at PNP related to 10 CFR 50, Appendix J are provided in this License Condition. Providing this License Condition is necessary to resume power operations.</td></tr>
</table>

License Condition 2.J	
Current License Condition 2.J [deleted]	Proposed License Condition 2.J ~~[deleted]~~ ***All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal scheduled, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.***

Basis

This License Condition is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this License Condition is reinstated to address the requirements of 10 CFR 50, Appendix H.

10 CFR 50 Appendix H requires that the design of the reactor vessel surveillance capsule program and withdrawal schedule must meet the requirements in the version of ASTM Standard Practice E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Code to which the reactor pressure vessel (RPV) was purchased. The rule also requires the licensee to perform capsule testing and to report the test results in accordance with the requirements in ASTM Standard Practice E 185-82 to the extent practicable for the configuration of the test specimens in the RPV surveillance capsules.

The requirements in Appendix H are applicable to nuclear plants that are in power operations in the reactor critical operating mode because: (a) this is the plant operating mode that produces high energy neutrons as a result of the reactor's nuclear fission process; and (b) the requirements are set in place to provide assurance that the RPV will maintain adequate levels of fracture toughness throughout the operating life of the reactor.

Continued implementation of the applicable surveillance capsule testing and reporting requirements are necessary for PNP because the plant will resume operation, thereby exposing the reactor vessel to high energy neutrons and subjected to high thermal stress environments, as induced by operating the reactor coolant system at an elevated temperature. Any corresponding commitments in the UFSAR, Revision 35, will also be reinstated under the provisions of 10 CFR 50.59.

License Condition 2.K	
<u>Current License Condition 2.K</u> This license is effective as of the date of issuance and until the Commission notifies the licensee in writing that the license is terminated.	<u>Proposed License Condition 2.K</u> This license is effective as of the date of issuance and ~~until the Commission notifies the licensee in writing that the license is terminated~~ ***shall expire at midnight March 24, 2031.***
Basis This License Condition is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this License Condition currently conforms with 10 CFR 50.51, *Continuation of license*, in that the license authorizes ownership and possession by HDI until the Commission notifies the licensee in writing that the license is terminated. While appropriate for a licensee that has permanently ceased operation, 10 CFR 50.51, *Continuation of license* requires that an operating reactor have a license issued for a fixed period of time. Therefore, this change is necessary to resume power operations.	

ATTACHMENTS AND DATE OF ISSUANCE	
<u>Current Attachments and Date of Issuance</u> Appendix A – Permanently Defueled Technical Specifications…	<u>Proposed Attachments and Date of Issuance</u> Appendix A – ~~Permanently Defueled~~ Technical Specifications…
Basis The title of Appendix A is updated to reflect that the Permanently Defueled Technical Specifications will be retitled as the Technical Specifications. This is an administrative change.	

3.1.2 Proposed Changes to the Permanently Defueled Technical Specifications

APPENDIX A TITLE PAGE	
Current Title PALISADES PLANT RENEWED FACILITY OPERATING LICENSE DPR-20 APPENDIX A PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS	Proposed Title PALISADES PLANT RENEWED FACILITY OPERATING LICENSE DPR-20 APPENDIX A ~~PERMANENTLY DEFUELED~~ TECHNICAL SPECIFICATIONS
Basis The title is modified to remove "Permanently Defueled" before "Technical Specifications" to reflect that PNP will be operating and not permanently defueled. This is an administrative change.	

APPENDIX A Table of Contents	
Current Table of Contents 1.0 USE AND APPLICATION… 5.7 High Radiation Area	deleted
Basis The Table of Contents is proposed to be removed from the PDTS and placed under licensee control. Placing the TOC under licensee control eliminates the regulatory burden of submitting TOC pages for NRC review and allows timely administrative corrections and improvements to the TOC without NRC review and approval. The TOC does not meet the criteria specified in 10 CFR 50.36 requiring its inclusion within a plant's TS. The TOC references the page number of each Specification in the TS and does not contain any technical information required by 10 CFR 50.36. Since the TOC does not include information required to be in the TS by 10 CFR 50.36, inclusion of a TOC within the TS is optional. Removal of the TOC from the TS is an administrative change and is acceptable. The TS TOC will be maintained, revised, and distributed in accordance with administrative procedures. Holders of copies of the TS, including the NRC, will continue to receive periodic updates of the TOC pages.	

TS SECTION 1.1, DEFINITIONS

TS 1.1, *Definitions*, provides defined terms that are applicable throughout the TS and TS Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AVERAGE DISINTEGRATION ENERGY $\bar{E}$	***$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the primary coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.***
AXIAL OFFSET (AO)	***AO shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core (determined using the incore monitoring system).***
AXIAL SHAPE INDEX (ASI)	***ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core (determined using the excore monitoring system).***
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training and retraining program required by Specification 5.3.2.
CHANNEL CALIBRATION	***A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST.*** ***Calibration of instrument channels with Resistance Temperature Detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.***

	Whenever a RTD or thermocouple sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. ***The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.***
CHANNEL CHECK	***A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.***
CHANNEL FUNCTIONAL TEST	***A CHANNEL FUNCTIONAL TEST shall be:*** ***a. Analog and bistable channels - the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY;*** ***b. Digital channels - the use of diagnostic programs to test (continued) digital hardware and the injection of simulated process data into the channel to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY.*** ***The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.***
CORE ALTERATION	***CORE ALTERATION shall be the movement of any fuel, sources, or control rods within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.***
CORE OPERATING LIMITS REPORT (COLR)	***The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation***

	within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	***DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1989; (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).***
INSERVICE TESTING PROGRAM	***The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).***
LEAKAGE	***LEAKAGE shall be:*** ***a. <u>Identified LEAKAGE</u>*** ***1. LEAKAGE, such as that from pump seals or valve packing (except Primary Coolant Pump seal water leakoff), that is captured and conducted to collection systems or a sump or collecting tank;*** ***2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; and*** ***3. Primary Coolant System (PCS) LEAKAGE through a Steam Generator to the Secondary System (primary to secondary LEAKAGE).*** ***b. <u>Unidentified LEAKAGE</u>*** ***All LEAKAGE (except Primary Coolant Pump seal leakoff) that is not identified LEAKAGE;*** ***c. <u>Pressure Boundary LEAKAGE</u>*** ***LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an PCS component body, pipe wall, or vessel wall.***
MODE	***A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average primary coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.***

NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 5.3.1.
OPERABLE – OPERABILITY	***A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).***
PHYSICS TESTS	***PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:*** ***a. Described in Chapter 13, Initial Tests and Operation, of the FSAR;*** ***b. Authorized under the provisions of 10 CFR 50.59; or*** ***c. Otherwise approved by the Nuclear Regulatory Commission.***
QUADRANT POWER TILT (Tq)	***Tq shall be the maximum positive ratio of the power generated in any quadrant minus the average quadrant power, to the average quadrant power.***
RATED THERMAL POWER (RTP)	***RTP shall be a total reactor core heat transfer rate to the primary coolant of 2565.4 MWt.***
REFUELING BORON CONCENTRATION	***REFUELING BORON CONCENTRATION shall be a Primary Coolant System boron concentration of > 1720 ppm and sufficient to assure the reactor is subcritical by > 5% Δρ with all control rods withdrawn.***
SHUTDOWN MARGIN (SDM)	***SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length control rods (shutdown and regulating) are fully inserted except for the single rod of highest reactivity worth, which is assumed to be fully withdrawn. However, with all full length control rods verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SOM calculation. With any full length control rods not capable of being fully inserted, the reactivity worth of these rods must be accounted for in the determination***

	of SDM; and b. There is no change in part length rod position.
STAGGERED TEST BASIS	***A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.***
THERMAL POWER	***THERMAL POWER shall be the total reactor core heat transfer rate to the primary coolant.***
TOTAL RADIAL PEAKING FACTOR (FRT)	***FRT shall be the maximum ratio of the individual fuel pin power to the core average pin power integrated over the total core height, including tilt.***

Table 1.1-1
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE PRIMARY COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 300
4	Hot Shutdown[(b)]	< 0.99	NA	$300 > T_{ave} > 200$
5	Cold Shutdown[(b)]	< 0.99	NA	≤ 200
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.
(b) All reactor vessel head closure bolts fully tensioned.
(c) One or more reactor vessel head closure bolts less than fully tensioned.

Basis This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, these Definitions are proposed to be reinstated because they are necessary to support the TS added by this LAR. When combined with the Definition for ACTIONS in the current PDTS, they comprise all the Definitions that were in the TS prior to Amendment 272 (Reference 8). Additionally, these Definitions have not changed from the Definitions used prior to PNP Amendment 272. Each Definition is used elsewhere in the proposed TS. The Definitions will be added to Section 1.0 in alphabetical order with the existing Definitions. The existing Definitions remain in the TS and are not modified by this LAR.

TS SECTION 1.2, LOGICAL CONNECTORS TS 1.2, *Logical Connectors*, explains how the arrangement of these connectors constitutes logical conventions with specific meanings. The proposed modifications reflect the logical connectors necessary to support the TS that are reinstated with this LAR. Modifications to Section 1.2 in the PDTS are shown below. The addition of previously deleted text is described here and shown in Attachment 1	
<u>Current PURPOSE</u> Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connector that appears in TS is <u>AND</u>. The physical arrangement of this connector constitutes logical conventions with specific meanings.	<u>Proposed PURPOSE</u> Logical connector***s*** that appear~~s~~ in TS ~~is~~ ***are*** <u>AND</u> ***and <u>OR</u>***. The physical arrangement of ~~this~~ ***these*** connector***s*** constitutes logical conventions with specific meanings.

Current BACKGROUND	Proposed BACKGROUND
Levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action).	***Several levels*** ~~Levels~~ of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). ***The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.*** ***When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.***
Current EXAMPLES The following example illustrates the use of logical connectors… EXAMPLE 1.2-1 REQUIRED ACTION A.1 Suspend ... AND A.2 Initiate ...	Proposed EXAMPLE~~S~~ The following example***s*** illustrate~~s~~ the use of logical connectors… EXAMPLE 1.2-1 REQUIRED ACTION A.1 ***Verify*** ~~Suspend~~ ... AND A.2 ***Restore*** ~~Initiate~~ ... EXAMPLE 1.2-2 is proposed to be reinstated. [*for entirety of text proposed for reinstatement, see Attachment 1*]

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this section will be reinstated. This section currently reflects the logical connector utilized in TS 3.7.14, TS 3.7.15, and TS 3.7.16. These are the only TS that utilize a logical connector in the PDTS. The Required Actions of Example 1.2-1 are currently "Suspend" and "Initiate" which more closely aligns with TS in the PDTS. They will be replaced with the standard "Verify" and "Restore." Reinstatement of this TS necessary to support power operation will include TS with the logical connector "OR" and more complex nesting. Therefore, Example 1.2-2 is reinstated since it pertains to the logical connector "OR" and more complex nesting. See Attachment 1 for the wording of Example 1.2-2. These changes are administrative.

TS SECTION 1.3, COMPLETION TIMES

TS 1.3, *Completion Times*, establishes the Completion Time convention, and provides guidance for its use. It is modified to reflect the power operation condition and the Completion Times that are proposed for the reinstated TS. Modifications to Section 1.3 are shown below. The addition of previously deleted text is described here and shown in Attachment 1.

Current BACKGROUND	Proposed BACKGROUND
Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring storage and handling of spent fuel.	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe ***operation of the plant*** ~~storage and handling of spent nuclear fuel~~.

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, the Background section of TS 1.3 is modified to reflect that PNP will be returned to power operation conditions. This change is administrative.

Current DESCRIPTION	Proposed DESCRIPTION
The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the discovery of a situation (e.g., variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. The Completion time begins when a Certified Fuel Handler (CFH) on the shift crew with responsibility for facility operations makes the determination that an LCO is not met and an ACTIONS Condition is entered. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the plant is not within the LCO Applicability.	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the discovery of a situation (e.g., ***inoperable equipment or*** variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the ***plant*** ~~facility~~ is in a ***MODE or*** specified condition stated in the Applicability of the LCO. ***Unless otherwise specified, the*** ~~The~~ Completion time begins when a ***senior licensed operator*** ~~Certified Fuel Handler (CFH)~~ on the ***operating*** shift crew with responsibility for plant operations makes the determination that an LCO is not met and an ACTIONS Condition is entered. ***The "otherwise specified***….[*for entirety of text proposed for reinstatement, see Attachment 1*]… ***in the Completion Time are satisfied.*** Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the plant is not within the LCO Applicability. ***If situations are discovered…***[*for entirety of text proposed for reinstatement, see Attachment 1*]… ***Example 1.3-3 may not be extended.***

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, the Description section of TS 1.3 is modified to reflect that PNP will be returned to a power operation condition. As a result, the TS will contain operability requirements for numerous systems and Completion Time rules are expanded to reflect the various situations in the reinstated TS. In addition, reinstatement of the term "plant" better represents PNP in the operating condition. "Senior licensed operator" and "operating shift crew" are reinstated to reflect the required operating staff. The terms "Certified Fuel Handler (CFH)" is deleted as these personnel will no longer have an operating role once these TS are reinstated. A separate LAR for administrative TS changes will address the staffing changes necessary to support a power operations plant.

Current EXAMPLES	Proposed EXAMPLE
The following example illustrates the use of Completion Times with different Required Actions. See Attachment 1 for complete text of EXAMPLE 1.3-1.	The following example***s*** illustrate~~s~~ the use of Completion Times with different ***types of Conditions and changing Conditions*** ~~Required Actions~~. EXAMPLE 1.3-1 is modified to address Completion Times as utilized by the reinstated TS. See Attachment 1 for the proposed changes. EXAMPLE 1.3-2 is proposed to be reinstated. EXAMPLE 1.3-3 is proposed to be reinstated. EXAMPLE 1.3-4 is proposed to be reinstated. EXAMPLE 1.3-5 is proposed to be reinstated EXAMPLE 1.3-6 is proposed to be reinstated. EXAMPLE 1.3-7 is proposed to be reinstated. See Attachment 1 for the entirety of the text to be reinstated.
Basis This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this section is modified to reflect the use of Completion Times that are utilized in the reinstated TS. The reinstated examples reflect the variety of conditions that are contained in the reinstated TS. These changes to the Examples section of TS 1.3 are administrative changes.	

TS SECTION 1.4, FREQUENCY TS 1.4, *Frequency*, defines the proper use and application of Frequency requirements. It is modified to reflect the reinstated TS which support the power operating condition for PNP. Modifications to Section 1.4 in the PDTS are shown below. The addition of previously deleted text is described here and shown in Attachment 1	
Current DESCRIPTION …The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR. The use of "met" and "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance	Proposed DESCRIPTION …The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR***, as well as certain Notes in the Surveillance column that modify performance requirements***. ***Sometimes special situations dictate when***

criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being “performed,” constitutes a Surveillance not "met.” “Performance” refers only to the requirement to specifically determine the ability to meet the acceptance criteria.	***the requirements of a Surveillance are to be met.*** … [See Attachment 1 for the text to be reinstated.] The use of "met” and “performed” in these instances conveys specific meanings. A Surveillance is "met” only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being “performed,” constitutes a Surveillance not "met.” “Performance” refers only to the requirement to specifically determine the ability to meet the acceptance criteria… [See Attachment 1 for the entirety of the text to be reinstated.] The use of "met” and “performed” in these instances conveys specific meanings. A Surveillance is "met” only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being “performed,” constitutes a Surveillance not "met.” ***Some Surveillances contain notes that modify the Frequency of (continued) performance or the conditions during which the acceptance criteria must be satisfied….***[See Attachment 1 for the entirety of the text to be reinstated.] ***Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.***

Current EXAMPLES	Proposed EXAMPLE
The following examples illustrate the type of Frequency statements that appear in the Technical Specifications (TS). Example 1.4-1… Example 1.4-2… [*for entirety of text for examples 1.4-1 and 1.4-2, see Attachment 1*]	The following examples ***illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3*** ~~illustrate the type of Frequency statement that appears in the Technical Specifications (TS)~~. Example 1.4-1 is modified to address Frequencies as utilized by the reinstated TS. See Attachment 1 for proposed changes. Example 1.4-2 is modified to address Frequencies as utilized by the reinstated TS. See Attachment 1 for proposed changes Example 1.4-3 is proposed to be reinstated. Example 1.4-4 is proposed to be reinstated. Example 1.4-5 is proposed to be reinstated. Example 1.4-6 is proposed to be reinstated.

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, the Description section of TS 1.4 is modified to reflect the reinstatement of the TS supporting a power operation condition. The number and types of Surveillance Requirements that are included in the operational TS contain many types of Frequencies and examples are added back into the TS to reflect these Frequencies. These changes to the Definitions and examples section of TS 1.4 are administrative changes.

TS SECTION 2.0, SAFETY LIMITS (SLs)

TS Section 2.0 contains SLs that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity from the reactor core and the primary coolant system (PCS) in accordance with 10 CFR 50.36(c)(1). The SLs established in TS Section 2.1 prevent overheating of the fuel and possible cladding perforation, which would result in the release of fission products to the reactor coolant and protects the integrity of the PCS from overpressurization, thereby preventing the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere. SL violations in TS Section 2.2 are values of various parameters for which automatic protective action is needed during normal operations or anticipated transients to prevent exceeding an SL.

Current TS 2.0	Proposed TS 2.0
TS 2.0 (Deleted)	TS 2.0 ***SAFETY LIMITS (SLs)*** ~~(Deleted)~~ ***TS 2.1 SLs…*** See Attachment 1 for the complete TS.

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS Section 2.0 is proposed for reinstatement since the SLs will apply to PNP reactor power operations in accordance with 10 CFR 50.36(c)(1).

TS 2.1.1, "Reactor Core SLs," prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. It is applicable in MODES 1 and 2. TS 2.1.1 applies to an operating reactor, and it will be reinstated.

TS 2.1.2, "Primary Cooling System (PCS) Pressure SL," protects the PCS from over-pressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere. It is applicable in MODES 1 through 6. TS 2.1.2 applies to maintaining the PCS pressure and it will be reinstated.

In 2.2, "SL Violations," TS 2.2.1 defines the action to take if SL 2.1.1 is not met. It requires the unit to be placed in MODE 3. TS 2.2.2 defines the action to take if SL 2.1.2 is not met. If the unit is in MODE 1 or 2, it requires the unit to be placed in MODE 3. If the unit is MODE 3, 4, 5, or 6, it requires compliance to be restored within five minutes. These TS will be reinstated.

These TS are required to comply with 10 CFR 50.36(c)(1).

TS SECTION 3.0, LIMITING CONDITIONS FOR OPERATION (LCO)

TS Section 3.0 contains the general requirements applicable to all Limiting Conditions for Operation (LCOs) and applies at all times unless otherwise stated in a TS. Proposed revisions to the PDTS (including those proposed for reinstatement) are described below. The corresponding TS Bases are also revised to reflect these changes.

A mark-up of this section is provided in Attachment 1.

Current LCO 3.0.1	Proposed LCO 3.0.1
LCOs shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2.	LCOs shall be met during the ***MODES or other*** specified conditions in the Applicability, except as provided in LCO 3.0.2***, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9.***

<table>
<tr><td colspan="2">Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, the MODES as defined in Table 1.1-1 apply to operating or refueling conditions and are used throughout the reinstated TS. Thus, the reference to MODES is reinstated. In addition, the references to LCOs 3.0.7, LCO 3.0.8, and 3.0.9 are reinstated to reflect the proposed reinstatement of those LCOs as discussed below.</td></tr>
<tr><td><u>Current LCO 3.0.2</u>

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.</td><td><u>Proposed LCO 3.0.2</u>

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.</td></tr>
<tr><td colspan="2">Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, LCO 3.0.2 is modified by reinstating the references to LCOs 3.0.5 and 3.0.6. This change reflects the proposed reinstatement of those LCOs as discussed below.</td></tr>
<tr><td><u>Current LCO 3.0.3</u>
Not included</td><td><u>Proposed LCO 3.0.3</u>
When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the plant shall be placed in a MODE or other specified condition in which the LCO is not applicable….LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.
For the entire text, see Attachment 1.</td></tr>
<tr><td colspan="2">Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, LCO 3.0.3 provides the actions that must be implemented when an LCO is not met. It is only applicable in MODES 1 through 4. These operating MODES are included in the reinstated TS. Thus, LCO 3.0.3 is needed to address these operational conditions.</td></tr>
</table>

<table>
<tr><td><u>Current LCO 3.0.4</u>
Not included</td><td><u>Proposed LCO 3.0.4</u>
When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:…This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the plant. For the entire text, see Attachment 1.</td></tr>
<tr><td colspan="2">Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, LCO 3.0.4 provides limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. MODES are included in the reinstated TS. Thus, LCO 3.0.4 is needed to address these operational conditions.</td></tr>
<tr><td><u>Current LCO 3.0.5</u>
Not included</td><td><u>Proposed LCO 3.0.5</u>
Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.</td></tr>
<tr><td colspan="2">Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, LCO 3.0.5 provides the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The allowance of LCO 3.0.5 to not comply with the requirements of LCO 3.0.2 (i.e., to not comply with the Required Actions) to allow the performance of SRs on equipment declared inoperable or removed from service is necessary as part of the reinstated TS because some TS will include requirements to declare equipment inoperable or to remove it from service. Thus, LCO 3.0.5 is needed to address these Required Actions.</td></tr>
</table>

Current LCO 3.0.6 Not included	Proposed LCO 3.0.6 ***When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered….When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.*** For the entire text, see Attachment 1.

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, LCO 3.0.6 addresses the actions required for a supported system when the support system LCO is not met. It is proposed to be reinstated since there are supported system LCOs in the reinstated TS. Thus, LCO 3.0.6 is needed to address these operational conditions.

Current LCO 3.0.7 Not included	Proposed LCO 3.0.7 ***Special Test Exception (STE) LCOs in each applicable LCO section allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.***

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, LCO 3.0.7 allows certain LCO exceptions when special tests are required to be performed at various times over the life of the plant. It is proposed for reinstatement since there are special test exceptions that will be reinstated in the TS. Thus, LCO 3.0.7 is needed to address these operational conditions.

<u>Current LCO 3.0.8</u> Not included	<u>Proposed LCO 3.0.8</u> ***When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:…At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.*** For the entire text, see Attachment 1.
Basis This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, LCO 3.0.8 addresses the actions required when one or more required snubbers are unable to perform their associated support function(s). It is proposed to be reinstated because, in an operating plant, snubbers are required to perform their associated support functions. Thus, LCO 3.0.8 is needed to address these operational conditions.	
<u>Current LCO 3.0.9</u> Not included	<u>Proposed LCO 3.0.9</u> ***When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed….At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.*** For the entire text, see Attachment 1.
Basis This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, LCO 3.0.9 addresses the actions required when one or more required barriers are unable to perform their related support function(s). It is proposed for reinstatement, because there are LCOs that require equipment to be operable or in operation in the TS. Barriers may be required to support certain TS functions. Thus, LCO 3.0.9 is needed to address these operational conditions.	

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TS SECTION 3.0, SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

TS Section 3.0 contains the general requirements applicable to all SRs and applies at all times unless otherwise stated in a TS. Proposed revisions to these PDTS are described below. The corresponding TS Bases are also revised to reflect these changes.

A mark-up of this section is provided.

</td></tr>
<tr><td>

<u>Current SR 3.0.1</u>

SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR…[*for entirety of text, see Attachment 1]…*Surveillances do not have to be performed on variables outside specified limits.

</td><td>

<u>Proposed SR 3.0.1</u>

SRs shall be met during the ***MODES or other*** specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR…[*for entirety of text, see Attachment 1]…*Surveillances do not have to be performed on ***inoperable equipment or*** variables outside specified limits.

</td></tr>
<tr><td colspan="2">

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this TS is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant.

SR 3.0.1 is modified by adding the reference to MODES. MODES are used in the reinstated TS. MODES as defined in Table 1.1-1 are for operating or refueling conditions. This term applies to a plant with reinstated TS.

In addition, SR 3.0.1 reinstates the discussion regarding inoperable equipment. The reinstated LCOs include equipment operability requirements.

</td></tr>
</table>

Current SR 3.0.2	Proposed SR 3.0.2
The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance.	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance ***or as measured from the time a specified condition of the Frequency is met***. ***For Frequencies specified as "once," the above interval extension does not apply.*** ***If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.*** ***Exceptions to this Specification are stated in the individual Specifications.***

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this TS is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant.

SR 3.0.2 provides an allowance for extending the frequency for performance of a SR to 1.25 times the interval specified in the Frequency to facilitate scheduling or unforeseen problems that may prevent performance during normal intervals. It is proposed to reinstate the discussion of frequency requirements that will exist in the reinstated TS LCOs.

Current SR 3.0.4	Proposed SR 3.0.4
Entry into a specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3.	Entry into a ***MODE or other*** specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. ***When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.*** ***This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the plant.***

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, this TS is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant.

SR 3.0.4 is modified by adding the reference to MODES. MODES are used in the reinstated TS. MODES as defined in Table 1.1-1 are for operating or refueling conditions. This term applies to a plant with reinstated TS.

In addition, SR 3.0.4 reinstates the provision that states that it shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The reinstated TS contain Required Actions that would require an entry into another specified condition defined in the Applicability of a TS.

TS SECTION 3.1, REACTIVITY CONTROL SYSTEMS

TS Section 3.1 contains requirements to assure and verify operability of reactivity control systems to ensure the reactor remains within the bounds of the PNP accident analyses.

TS Section 3.1 is proposed for reinstatement in its entirety. Thus, a mark-up of this TS section is not provided. See Attachment 2 for the complete text of the reinstated TS. The corresponding TS Bases are also reinstated to reflect these changes.

Proposed PNP TS	Basis for Change
TS 3.1.1, SHUTDOWN MARGIN (SDM)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.1.1 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.1.1 ensures the SDM is maintained within the limits specified in the COLR. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown events and Anticipated Operational Occurrences (AOOs). The SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.1.2, Reactivity Balance	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.1.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.1.2 ensures that core reactivity balance remains within ± 1% of predicted values. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown events and Anticipated Operational Occurrences (AOOs). The SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.1.3, Moderator Temperature Coefficient (MTC)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to

	10 CFR 50.82(a)(2), Reference 3, TS 3.1.3 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.1.3 establishes MTC limits during plant operation to ensure stable plant operation. The MTC relates a change in core reactivity to a change in primary coolant temperature. The MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.1.4, Control Rod Alignment	This TS is proposed for reinstatement in its entirety, with revision, to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.1.4 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant, except for the NOTE in SR 3.1.4.3. Maximum control rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Control rod alignment satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2). The NOTE in SR 3.1.4.3 excepted a control rod drive from the SR for a specific period of time. The cycle referenced in the NOTE has passed and all control rod drives are subject to the SR going forward. Therefore, the NOTE is no longer needed and is not reinstated. This NOTE is related to previous License Condition 2.C.(4) which was deleted as a historical License Condition in Reference 3. See Reference 3, section 4.2.11 for the discussion.
TS 3.1.5, Shutdown and Part-Length Control Rod Group Insertion Limits	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.1.5 is reinstated as it existed in the previously

	approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.1.5 establishes limits on insertion of PNP's shutdown and part length control rods to ensure that core reactivity, ejected rod worth, and SDM are preserved. Limits on shutdown rod insertion have been established, and all rod positions are monitored and controlled during power operation. The shutdown and part-length rod group insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.1.6, Regulating Rod Group Position Limits	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.1.6 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.1.6 ensures regulating rod groups are limited to the withdrawal sequence, overlap, and insertion limits specified in the COLR. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The regulating rod group position limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.1.7, Special Test Exceptions (STE)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.1.7 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.1.7 permits suspension of existing LCOs to allow the performance of certain PHYSICS TESTS in MODE 2. These tests are conducted to determine control rod worths, SHUTDOWN MARGIN (SDM), and specific reactor core characteristics. It is acceptable to suspend certain

	LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded.

TS SECTION 3.2, POWER DISTRIBUTION LIMITS
TS Section 3.2 contains power distribution limits that provide assurance that fuel design criteria are not exceeded, and the accident analysis assumptions remain valid. TS Section 3.2 is proposed for reinstatement in its entirety. Thus, a mark-up of this TS section is not provided. See Attachment 2 for the complete text of the reinstated TS. The corresponding TS Bases are also reinstated to reflect these changes.

Proposed PNP TS	Basis for Change
TS 3.2.1, Linear Heat Rate (LHR)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.2.1 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.2.1 ensures the LHR remains within the limits specified by the COLR. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient. LHR satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.2.2, Total Radial Peaking Factor (F_R^T)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.2.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.2.2 ensures the F_R^T remains within the limits specified in the COLR. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating

	within acceptable bounding conditions at the onset of a transient. The Total Radial Peaking Factor satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.2.3, Quadrant Power Tilt (T_q)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.2.3 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The limitations on T_q, specified in TS 3.2.3, ensure that assumptions used in the analysis for establishing LHR limits and DNB margin remain valid during operation. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient. The T_q satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.2.4, Axial Shape Index (ASI)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.2.4 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.2.4 ensures the ASI remains within the limits specified in the COLR. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient. The ASI satisfies Criterion 2 of 10 CFR 50.36(c)(2).

TS SECTION 3.3, INSTRUMENTATION

TS Section 3.3 contains operability requirements for sensing and control instrumentation required for safe operation of the facility.

TS Section 3.3 is proposed for reinstatement in its entirety. Thus, a mark-up of this TS section is not provided. See Attachment 2 for the complete text of the reinstated TS. The corresponding TS Bases are also reinstated to reflect these changes.

Proposed PNP TS	Basis for Change
TS 3.3.1, Reactor Protection System (RPS) Instrumentation	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.3.1, including Tables 3.3.1-1 and 3.3.1-2, is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The RPS initiates a reactor shutdown as required to mitigate design basis accidents and transients. By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents. The RPS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.3.2, Reactor Protective System (RPS) Logic and Trip Initiation	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.3.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The RPS initiates a reactor shutdown as required to mitigate design basis accidents and transients. By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents. The RPS Logic and Trip Initiation satisfy Criterion 3 of 10 CFR 50.36(c)(2).

TS 3.3.3, Engineered Safety Features (ESF) Instrumentation	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.3.3, including Table 3.3.3-1, is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. 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 Instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.3.4, Engineered Safety Features (ESF) Logic and Manual Initiation	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.3.4, including Table 3.3.4-1, is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. 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. ESF manual initiation permits the operator to manually actuate an ESF system when necessary. The ESF satisfies Criterion 3 of 10 CFR 50.36(c)(2).

T 3.3.5, Diesel Generator (DG) – Undervoltage Start (UV Start) Instrumentation	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.3.5 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The DGs provide a source of emergency power to allow safe operation of the plant. Undervoltage protection instrumentation will generate a DG start in the event a loss of voltage or degraded voltage condition occurs. The DG - UV Start channels satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.3.6, Refueling Containment High Radiation (CHR) Instrumentation	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.3.6 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. CHR Instrumentation provides automatic containment isolation during refueling operations. This ensures the radioactive materials are not released directly to the environment and significantly reduces the offsite doses from those calculated by the safety analyses. The Refueling CHR Instrumentation satisfies the requirements of Criterion 4 of 10 CFR 50.36(c)(2).
TS 3.3.7, Post Accident Monitoring (PAM) Instrumentation	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.3.7, including Table 3.3.7-1, is reinstated as it existed in the previously approved TS prior to Amendment

	272, to reflect the power operation condition of the plant. The PAM instrumentation displays plant variables that provide information required by the operators during accident situations. This information provides the necessary support for the operator to take the manual actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety Functions for Design Basis Events. PAM instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 meets Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.3.8, Alternate Shutdown System	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.3.8, including Table 3.3.8-1, is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The Alternate Shutdown System provides the control room operator with sufficient instrumentation and controls to maintain the plant in a safe shutdown condition from a location other than the control room. The Alternate Shutdown System has been identified as an important contributor to the reduction of plant risk to accidents and, therefore, satisfies the requirements of Criterion 4 of 10 CFR 50.36(c)(2).
TS 3.3.9, Neutron Flux Monitoring Channels	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.3.9 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant.

	The neutron flux monitoring channels are necessary to monitor core reactivity changes. By monitoring neutron flux, loss of SDM caused by boron dilution can be detected as an increase in flux. The neutron flux monitoring channels satisfy Criterion 4 of 10 CFR 50.36(c)(2).
TS 3.3.10 Engineered Safeguards Room Ventilation (ESRV) Instrumentation	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.3.10 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The ESRV instrumentation provides isolation of the engineered safeguards pump rooms in the event of high radiation in the pump rooms. Typically, high radiation would only be expected due to excessive leakage during the recirculation phase of operation following a Loss of Coolant Accident (LOCA). The ESRV Instrumentation satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2).

TS SECTION 3.4, PRIMARY COOLANT SYSTEM (PCS)

TS Section 3.4 contains requirements that provide for appropriate control of process variables, design requirements, or operating restrictions needed for appropriate functional capability of PCS equipment required for safe operation of the facility.

TS Section 3.4 is proposed for reinstatement in its entirety. Thus, a mark-up of this TS section is not provided. See Attachment 2 for the complete text of the reinstated TS. The corresponding TS Bases are also reinstated to reflect these changes.

Proposed PNP TS	Basis for Change
TS 3.4.1, PCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.1 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The requirements of TS 3.4.1 ensure that PCS pressure, temperature and flow rate remain within the limits specified in the COLR. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses. The PCS DNB limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.4.2, PCS Minimum Temperature for Criticality	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.2 ensures the PCS temperature remains above the minimum temperature for reactor criticality to prevent operation in an unanalyzed condition. The PCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.4.3, PCS Pressure and Temperature (P/T) Limits	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.3, including Figures 3.4.3-1 and 3.4.3-2, is reinstated as it existed in the previously approved TS prior to

	Amendment 272, to reflect the power operation condition of the plant. TS 3.4.3 limits the pressure and temperature changes during PCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation. The PCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.4.4, PCS Loops – MODES 1 and 2	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.4 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.4 ensures adequate PCS heat transfer capability during power operation. 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. PCS Loops - MODES 1 and 2 satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2).
TS 3.4.5, PCS Loops – MODE 3	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.5 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.5 ensures adequate PCS heat transfer capability during hot standby. 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.

	PCS Loops - MODE 3 satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.4.6, PCS Loops – MODE 4	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.6 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.6 ensures adequate PCS heat transfer capability during hot shutdown. The intent of this LCO is to provide forced flow for decay heat removal and transport. PCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2).
TS 3.4.7, PCS Loops – MODE 5, Loops Filled	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.7 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.7 ensures adequate PCS heat transfer capability during cold shutdown with the PCS piping filled. 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 or the Shutdown Cooling (SDC) heat exchangers. PCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2).

TS 3.4.8, PCS Loops – MODE 5, Loops Not Filled	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.8 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.8 ensures adequate PCS heat transfer capability during cold shutdown with the PCS piping not filled. In MODE 5 with loops not filled, only the SDC System can be used for coolant circulation. PCS loops - MODE 5 (Loops Not Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2).
TS 3.4.9, Pressurizer	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.9 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.9 provides requirements for pressurizer water level, required pressurizer heater capacity, and heater power supply to ensure proper operation of the pressurizer. The pressurizer provides a point in the PCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes. The pressurizer satisfies Criterion 2 (for pressurizer water level) and Criterion 4 (for pressurizer heaters) of 10 CFR 50.36(c)(2).
TS 3.4.10, Pressurizer Safety Valves	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.10, including Table 3.4.10-1, is reinstated as it existed

	in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.10 ensures the pressurizer safety valves are capable of providing PCS overpressure protection. Operating in conjunction with the Reactor Protection System, these valves are used to ensure that the pressure Safety Limit is not exceeded for analyzed transients. The pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.4.11, Pressurizer Power Operated Relief Valves (PORVs)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.11 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.11 ensures the PORVs are capable of providing PCS overpressure protection. The primary purpose of this LCO is to ensure that the PORV and the block valve are operating correctly so the potential for a LOCA through the PORV pathway is minimized, or if a LOCA were to occur through a failed open PORV, the block valve could be manually operated to isolate the path. Pressurizer PORVs satisfy Criterion 4 of 10 CFR 50.36(c)(2).

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.12, including Figure 3.4.12-1, is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.12 ensures the integrity of the primary coolant pressure boundary by maintaining the PCS pressure within allowable values (the Pressure and Temperature (P/T) limits of 10 CFR 50, Appendix G) at low temperatures. The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.4.13, PCS Operational LEAKAGE	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.13 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.13 limits operation when PCS leakage is present. The purpose of the PCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from certain sources to amounts that do not compromise safety. PCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2).

TS 3.4.14, PCS Pressure Isolation Valve (PIV) Leakage	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.14 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.14 ensures that PIV leakage or inadvertent valve positioning does not result in overpressure of low-pressure piping and components. Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low-pressure portions of connecting systems. PCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.4.15, PCS Leakage Detection Instrumentation	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.15 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.15 ensures instrumentation is provided to detect and identify PCS leakage. Leakage detection instrumentation must have the capability to detect significant Primary Coolant Pressure Boundary (PCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. PCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2).

TS 3.4.16, PCS Specific Activity	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.16 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.16 limits the allowable concentration level of radionuclides in the primary coolant to minimize the offsite dose consequences in the event of a steam generator tube rupture or other accident. PCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.4.17, Steam Generator (SG) Tube Integrity	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.4.17 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.4.17 ensures the primary containment pressure boundary (PCPB) function of the SG. This Specification addresses only the PCPB integrity function of the SG. Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

TS SECTION 3.5, EMERGENCY CORE COOLING SYSTEMS (ECCS)

TS Section 3.5 contains requirements that provide for appropriate functional capability of ECCS equipment required for mitigation of DBAs or transients to protect the integrity of a fission product barrier.

TS Section 3.5 is proposed for reinstatement in its entirety. Thus, a mark-up of this TS section is not provided. See Attachment 2 for the complete text of the reinstated TS. The corresponding TS Bases are also reinstated to reflect these changes.

Proposed PNP TS	Basis for Change
TS 3.5.1, Safety Injection Tanks (SITs)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.5.1 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.5.1 ensures the functions of the SITs are maintained. They are to supply water to the reactor vessel during the blowdown phase of a Loss of Coolant Accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Primary Coolant System (PCS) makeup for a small break LOCA. The SITs satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.5.2, ECCS – Operating	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.5.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.5.2 ensures the ECCS can provide core cooling and negative reactivity to protect the reactor core during accidents involving inventory loss. ECCS - Operating satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.5.3, ECCS – Shutdown	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.5.3 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power

	operation condition of the plant. TS 3.5.3 ensures the ECCS can provide core cooling and negative reactivity to protect the reactor core during accidents involving inventory loss while in the hot shutdown condition. ECCS - Shutdown satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.5.4, Safety Injection Refueling Water Tank (SIRWT)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.5.4 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.5.4 ensures a source of borated water is available for containment spray system and engineered safeguards pump operation. The SIRWT satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.5.5, Containment Sump Buffering Agent and Weight Requirements	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.5.5 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.5.5 ensures the buffering agent, i.e., sodium tetraborate, results in a post-LOCA sump water pH value consistent with accident analyses. STB satisfies Criterion 3 of 10 CFR 50.36(c)(2).

TS SECTION 3.6, CONTAINMENT SYSTEMS	
TS Section 3.6 contains requirements that assure the integrity of the containment, depressurization and cooling systems, and containment isolation valves. TS Section 3.6 is proposed for reinstatement in its entirety. Thus, a mark-up of this TS section is not provided. See Attachment 2 for the complete text of the reinstated TS. The corresponding TS Bases are also reinstated to reflect these changes.	
Proposed PNP TS	Basis for Change
TS 3.6.1, Containment	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.6.1 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.6.1 ensures a containment configuration that is consistent with the safety analyses. Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.6.2, Containment Air Locks	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.6.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.6.2 ensures the containment air locks perform their design function as part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to limiting the containment leakage rate to $\leq 1.0\ L_a$.

	The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.6.3, Containment Isolation Valves	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.6.3 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.6.3 ensures the containment isolation valves and devices are capable of providing containment isolation within the time limits assumed in the safety analyses. The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.6.4, Containment Pressure	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.6.4 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.6.4 ensures containment pressure is limited during normal operation to preserve the initial conditions assumed in accident analyses. Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.6.5, Containment Air Temperature	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.6.5 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.6.5 ensures containment pressure is limited during normal operation to preserve the initial

	conditions assumed in accident analyses. Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2).
TS 3.6.6, Containment Cooling Systems	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.6.6 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.6.6 ensures Containment Spray and Containment Air Cooler systems are capable of limiting post-accident pressure and temperature in containment to less than the design values. The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2).

TS SECTION 3.7, PLANT SYSTEMS

TS Section 3.7 provides requirements for the appropriate functional capability of plant equipment required for safe operation of the facility, including requirements that apply when the facility is in a defueled condition.

TS 3.7.1 through TS 3.7.13, and TS 3.7.17 are proposed for reinstatement in their entirety. Thus, a mark-up of these TS is not provided. See Attachment 2 for the complete text of the reinstated TS.

TS 3.7.14 through TS 3.7.16 are modified to reinstate the reference to LCO 3.0.3. PNP will also use the Surveillance Frequency Control Program (SFCP), implemented in Amendment 271 (Reference 10), in the TS. Therefore, the Frequency of SR 3.7.14.1 and SR 3.7.15.1 will revert to the SFCP. The actual values of the Frequency do not change. The relocation of the Frequencies is administrative in nature.

Mark-ups of TS 3.7.14, TS 3.7.15, and TS 3.7.16 are provided in Attachment 1 in this enclosure.

The corresponding TS Bases are also reinstated and revised to reflect these changes.

Proposed PNP TS	Basis for Change
TS 3.7.1, Main Steam Safety Valves (MSSVs)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of

	the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.1, including Table 3.7.1-1, is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.1 ensures the MSSVs are capable of providing protection against over-pressurization of the secondary system and the primary coolant system. The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.2, Main Steam Isolation Valves (MSIVs)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.2 ensures the MSIVs are capable of isolating steam flow from the secondary side of the steam generators in the event of a high energy line break. The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.3, Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.3 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.3 ensures the MFRVs and MFRV bypass valves provide steam generator level control during normal plant operation and provide isolation in the event of a high energy line break.

	The MFRVs and MFRV bypass valves satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.4, Atmospheric Dump Valves (ADVs)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.4 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.4 ensures the ADVs are capable of removing decay heat should the preferred heat sink not be available. The ADVs satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.5, Auxiliary Feedwater (AFW) System	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.5 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.5 ensures the AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Primary Coolant System upon the loss of normal feedwater supply. The AFW System satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.6, Condensate Storage and Supply	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.6 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant.

	TS 3.7.6 ensures a supply of 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 and Supply satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.7, Component Cooling Water (CCW) System	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.7 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.7 ensures a heat sink for the removal of process and operating heat from safety related components during a DBA or transient. The CCW System satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.8, Service Water System (SWS)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.8 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.8 ensures a heat sink for the removal of process and operating heat from safety related components during a DBA or transient. The SWS satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.9, Ultimate Heat Sink (UHS)	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.9 is

	reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.9 ensures a heat sink is provided for process and operating heat from safety related components during a DBA or transient, as well as during normal operation. The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.10, Control Room Ventilation (CRV) Filtration	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.10 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.10 ensures the CRV Filtration system provides a protected environment from which occupants can control the plant following an uncontrolled release of radioactivity. The CRV Filtration system satisfies Criterion 3 of 10 CFR 50.36(c)(2).
3.7.11, Control Room Ventilation (CRV) Cooling	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.11 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.11 ensures the CRV Cooling system provides temperature control for the control room during normal and emergency conditions. The CRV Cooling system satisfies Criterion 3 of 10 CFR 50.36(c)(2).

3.7.12, Fuel Handling Area Ventilation System	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.12 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.12 ensures the Fuel Handling Area Ventilation System filters airborne radioactive particulates from the area of the spent fuel pool following a fuel handling accident or a fuel cask drop accident. For the fuel handling accident, the Fuel Handling Area Ventilation System satisfies Criterion 4 of 10 CFR 50.36(c)(2). For the fuel cask drop accident, the Fuel Handling Area Ventilation System satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.13, Engineered Safeguards Room Ventilation (ESRV) Dampers	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.13 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The ESRV dampers provide isolation of the engineered safeguards room in the event of a high radiation alarm. The ESRV Dampers satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.7.14, Spent Fuel Pool (SFP) Water Level	TS 3.7.14 is in the PDTS. Modifications to this TS are shown in Attachment 1. These modifications ensure that TS 3.7.14 is reinstated in its entirety, as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.14 the initial SFP water level assumed in the fuel handling accident analysis and cask drop analysis is maintained.

	The TS section title is administratively changed from "Facility Systems" to "Plant Systems" to reflect that this TS section addresses operational facility requirements. In addition, the NOTE in the Actions ("LCO 3.0.3 is not applicable") is proposed to be reinstated to conform to the reinstatement of TS LCO 3.0.3 as previously proposed. PNP will use the SFCP, implemented as Amendment 271 (Reference 10), in the TS. Therefore, the Frequency of SR 3.7.14.1 will revert to the SFCP. Consequently, SR 3.7.14.1 Frequency is changed from “7 days” to "In accordance with the Surveillance Frequency Control Program". The actual value of the Frequency does not change. The relocation of this Frequency is administrative in nature.

TS 3.7.15, Spent Fuel Pool (SFP) Boron Concentration	TS 3.7.15 is in the PDTS. Modifications to this TS are shown in Attachment 1. These modifications ensure that TS 3.7.15 is reinstated in its entirety, as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.15 ensures the initial SFP boron concentration assumed in the fuel handling accident analysis in maintained. The TS section title is administratively changed from "Facility Systems" to "Plant Systems" to reflect that this TS section addresses operational facility requirements. In addition, the NOTE in the Actions ("LCO 3.0.3 is not applicable") is proposed to be reinstated to conform to the reinstatement of TS LCO 3.0.3 as previously proposed. PNP will use the SFCP, implemented as Amendment 271 (Reference 10), in the TS. Therefore, the Frequency of SR 3.7.15.1 will revert to the SFCP. Consequently, SR 3.7.15.1 Frequency is changed from “7 days” to "In accordance with the Surveillance Frequency Control Program". The actual value of the Frequency does not change. The relocation of this Frequency is administrative in nature.
TS 3.7.16, Spent Fuel Pool Storage	TS 3.7.16, including Tables 3.7.16-1 through 3.7.16-5, is in the PDTS. Modifications to this TS are shown in Attachment 1. These modifications ensure that TS 3.7.16 is reinstated in its entirety, as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The TS section title is administratively changed from "Facility Systems" to "Plant Systems" to reflect that this TS section addresses operational facility requirements. In addition, the NOTE in the Actions ("LCO 3.0.3 is not applicable") is proposed to be reinstated to conform to the reinstatement of TS LCO 3.0.3 as previously proposed.

TS 3.7.17, Secondary Specific Activity	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.7.17 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.7.17 ensures steam generator tube out-leakage is identified. A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents. Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

TS SECTION 3.8, ELECTRICAL POWER SYSTEMS

TS Section 3.8 contains operability requirements that provide for appropriate functional capability of plant electrical equipment required for safe operation of the facility.

TS Section 3.8 is proposed for reinstatement in its entirety. Thus, a mark-up of this TS section is not provided. See Attachment 2 for the complete text of the reinstated TS. The corresponding TS Bases are also reinstated to reflect these changes.

Proposed PNP TS	Basis for Change
TS 3.8.1, AC Sources – Operating	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.8.1 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.8.1 ensures independent and redundant sources of power to the ESF systems comprised of offsite power sources and onsite standby diesel generators. The AC sources satisfy Criterion 3 of

	10 CFR 50.36(c)(2).
TS 3.8.2, AC Sources – Shutdown	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.8.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.8.2 ensures independent and redundant sources of power to the ESF systems comprised of offsite power sources and onsite standby diesel generators. The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.8.3, Diesel Fuel, Lube Oil, and Starting Air	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.8.3 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.8.3 ensures the diesel generators (DGs) are capable of performing their design function. Since diesel fuel, lube oil, and starting air subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.8.4, DC Sources – Operating	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.8.4 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant.

	TS 3.8.4 ensures the DC electrical system supports the AC power system and selected safety related equipment. The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.8.5, DC Sources – Shutdown	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.8.5 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.8.5 ensures the DC electrical system supports the AC power system and selected safety related equipment. The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.8.6, Battery Cell Parameters	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.8.6, including Table 3.8.6-1, is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.8.6 ensures battery cell parameters remain within acceptable limits to ensure availability of the required DC power. Battery cell parameters satisfy Criterion 3 of 10 CFR 50.36(c)(2).

TS 3.8.7, Inverters – Operating	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.8.7 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.8.7 ensures the inverters are capable of providing continuous AC power to the preferred AC buses. Inverters are part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.8.8, Inverters – Shutdown	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.8.8 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.8.8 ensures the inverters are capable of providing continuous AC power to the preferred AC buses. Inverters are part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.8.9, Distribution Systems – Operating	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.8.9 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.8.9 ensures an independent and redundant source of power to the ESF systems.

	The distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.8.10, Distribution Systems – Shutdown	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.8.10 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.8.10 ensures an independent and redundant source of power to the ESF systems. The distribution system satisfies Criterion 3 of 10 CFR 50.36(c)(2).

TS SECTION 3.9, REFUELING OPERATIONS

TS Section 3.9 contains requirements that provide for appropriate functional capability of parameters and equipment that are required for mitigation of DBAs during refueling operations (moving irradiated fuel to or from the reactor core).

TS Section 3.9 is proposed for reinstatement in its entirety. Thus, a mark-up of this TS section is not provided. See Attachment 2 for the complete text of the reinstated TS. The corresponding TS Bases are also reinstated to reflect these changes.

Proposed PNP TS	Basis for Change
TS 3.9.1, Boron Concentration	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.9.1 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.9.1 ensures the boron concentration of the PCS and the refueling cavity during refueling activities remain within limits to ensure that the reactor remains subcritical during MODE 6. Boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2).

TS 3.9.2, Nuclear Instrumentation	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.9.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.9.2 ensures instrumentation is available to monitor the core reactivity condition during refueling activities. Nuclear Instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2).
TS 3.9.3, Containment Penetrations	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.9.3 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.9.3 ensures that a release of fission product radioactivity from a fuel handling accident in containment will be mitigated. The Containment Penetrations satisfy the requirements of Criterion 4 of 10 CFR 50.36(c)(2).

TS 3.9.4, Shutdown Cooling (SDC) and Coolant Circulation – High Water Level	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.9.4 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.9.4 ensures the removal of decay and sensible heat and the mixing of borated coolant during refueling activities. SDC and Coolant Circulation - High Water Level satisfies Criterion 4 of 10 CFR 50.36(c)(2).
TS 3.9.5, Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.9.5 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.9.5 ensures the removal of decay and sensible heat and the mixing of borated coolant during refueling activities. SDC and Coolant Circulation - Low Water Level satisfies Criterion 4 of 10 CFR 50.36(c)(2).
TS 3.9.6, Refueling Cavity Water Level	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 3.9.6 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 3.9.6 ensures sufficient water level in the refueling cavity and spent fuel pool during refueling activities.

	Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2).

TS Section 4.0, DESIGN FEATURES	
TS Section 4.0, Design Features, provides information and design requirements associated with plant systems. TS 4.2 and TS 4.3.1.4 apply in an operating condition and are proposed to be reinstated. See Attachment 1 for the marked-up pages.	
Proposed PNP TS	Basis for Change
TS 4.2, Reactor Core ***4.2.1 <u>Fuel Assemblies</u>*** ***The reactor core shall contain 204 fuel assemblies. Each assembly shall consist of a matrix…***[for entirety of text proposed for reinstatement, see Attachment 1]. ***4.2.2 <u>Control Rod Assemblies</u>*** ***The reactor core shall contain 45 control rods. Four of these control rods may consist of part-length absorbers. The control material shall be silver-indium-cadmium, as approved by the NRC.***	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 4.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 4.2 provides requirements for reactor fuel assemblies and control rods in the reactor core. In an operating plant, fuel assemblies and control rod requirements for the core are necessary. Therefore, this TS will be reinstated.
4.3.1 <u>Criticality</u> 4.3.1.1.a ***New or i***rradiated fuel assemblies… 4.3.1.2.e ***New or i***rradiated fuel assemblies… 4.3.1.3.e ***New or i***rradiated fuel assemblies… *[For entirety of TS 4.3.1 text, see Attachment 1]*	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 4.3.1 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. These TS sections are proposed to be revised to reinstate references to new fuel storage. TS Sections 4.3.1.1, 4.3.1.2, and 4.3.1.3 provide requirements to ensure the new and irradiated fuel stored in racks in

	Regions I and II of the spent fuel pool remains subcritical. As an operating plant, PNP will receive new fuel and these sections address storage of that new fuel.
TS 4.3.1.4 ***The new fuel storage racks are designed and shall be maintained with***…[for entirety of text proposed for reinstatement, see Attachment 1].	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 4.3.1.4, including Figure 4.3-1, is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 4.3.1.4 provides requirements for the fuel racks for new fuel assemblies. With the return to power operation, PNP will be authorized to receive new fuel assemblies for reactor reloads.

TS Section 5.0, Administrative Controls	
TS Section 5.0 establishes the requirements associated with site personnel responsibilities, the site organization, staffing, training, procedures, programs, reporting requirements, and high radiation areas. Portions of this section are proposed to be reinstated as administrative requirements are needed for power operation of PNP. Sections reinstated in their entirety are shown in Attachment 2. Sections that are marked up from the current PDTS are shown in Attachment 1.	
Proposed PNP TS	Basis for Change
TS 5.5.2, <u>Primary Coolant Sources Outside Containment</u>	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.2 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. This program provides controls to minimize primary coolant leakage to the

	engineered safeguards rooms from portions of systems outside containment during mitigation of a DBA occurring in containment. Those systems include the containment spray system, the safety injection system, the shutdown cooling system, and the containment sump suction piping. As discussed in Section 3.0, the UFSAR, Revision 35, Chapter 14 accidents and transients inside containment are applicable and therefore, recirculation of post-accident highly radioactive primary coolant outside containment could occur. Therefore, this program is reinstated in the TS.
TS 5.5.5, <u>Containment Structural Integrity Surveillance Program</u>	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.5 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. This program controls monitoring of several containment attributes to ensure containment structural integrity. During power operations the containment is credited as part of the initial conditions of the applicable accident analyses and as part of the primary success path for mitigation of these events. In addition, this program is invoked in TS 3.6.1, “Containment” which is being reinstated. Therefore, this program is also reinstated in the TS.
TS 5.5.6, <u>Primary Coolant Pump Flywheel Surveillance Program</u>	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the

	exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.6 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. With the return to power operation, the flywheel surveillance program provides the inspection frequencies and acceptance criteria for the reactor coolant pump flywheel inspection program. Therefore, this program is reinstated in the TS.
TS 5.5.8, Steam Generator (SG) Program	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.8 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. This TS requires that a steam generator program be established and implemented to ensure that steam generator tube integrity is maintained. With the reinstatement of TS 3.4.17, “Steam Generator Tube Integrity” this program is needed as this TS invokes its use. Therefore, this program is also reinstated in the TS.
TS 5.5.9, Secondary Water Chemistry Program	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.9 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The purpose of the secondary water chemistry program is to maintain water chemistry to inhibit steam generator tube

	degradation. This program is necessary for the operating plant. Therefore, this program is reinstated in the TS.
TS 5.5.10, Ventilation Filter Testing Program	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.10 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. This program implements certain testing procedures for the control room ventilation and fuel handling area ventilation systems. With the reinstatement of TS 3.7.10, "Control Room Ventilation (CRV) Filtration" and TS 3.7.12, "Fuel Handling Area Ventilation" this program is needed as these TS invoke its use. Therefore, this program is also reinstated in the TS.
TS 5.5.11, Fuel Oil Testing Program	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.11 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. This program implements required testing of both new fuel oil and stored fuel oil to ensure compliance with manufacturer's specifications and applicable ASTM Standards. With the reinstatement of TS 3.8.3, "Diesel Fuel, Lube Oil and Starting Air," this program is needed as this TS invokes its use. Therefore, this program is also reinstated in the TS.

TS 5.5.13, <u>Safety Functions Determination Program (SFDP)</u>	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.13 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. This program was established to ensure loss of safety function is detected and appropriate actions taken when a supported system LCO is not met solely due to a support system LCO not being met. With the resumption of power operation and reinstatement of the TS, redundant systems are required to mitigate the UFSAR, Revision 35, Chapter 14 accidents and transients. Therefore, the requirements of the SFDP, which directs cross train checks of multiple and redundant safety systems, apply. Additionally, the SFDP is invoked in LCO 3.0.6, which is being reinstated in its entirety as previously discussed. Therefore, this program is also reinstated in the TS.
TS 5.5.14, <u>Containment Leak Rate Testing Program</u>	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.14 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 5.5.14 requires that a program to address the leakage rate testing of the containment be established. With the resumption of power operation and the reinstatement of the TS, Containment

	integrity is credited in the analysis of the UFSAR, Revision 35, Chapter 14 accidents and transients. In addition, this program is invoked in TS 3.6.1, “Containment” and TS 3.6.2, “Containment Air Locks” which are being reinstated. Therefore, this program is also reinstated in the TS.
TS 5.5.16, Control Room Envelope Habitability Program	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.16 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 5.5.16 establishes the administrative program for testing of the control room habitability systems to ensure operators can safely implement actions to control the reactor and mitigate accidents from within the control room envelope. These tests of control room habitability systems are necessary to support safe operation of the plant because reactor accidents challenging control room habitability are possible. With the reinstatement of TS 3.7.10, "Control Room Ventilation (CRV) Filtration" this program is needed as this TS invokes its use. Therefore, this program is also reinstated in the TS.
TS. 5.5.17, Surveillance Frequency Control Program	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.5.17 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant.

	The SFCP at PNP was implemented via Amendment 271 (Reference 12). The SFCP requires that a program be established to ensure that SRs specified in the TS are performed at intervals sufficient to assure the associated LCOs are met. PNP uses the SFCP in the TS as shown in Attachment 2. With the reinstatement of numerous TS invoking its use, this program is also reinstated.
TS 5.6.2, <u>Radiological Environmental Operating Report</u>	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.6.2 is modified by replacing the reference to the "facility" with the "plant" to reflect the power operation condition as follows: "The Radiological Environmental Operating Report covering the operation of the ***plant*** ~~facility~~ during…[for entirety of 5.6.2 text, see Attachment 1]…" This is an administrative change.
TS 5.6.3, <u>Radioactive Effluent Release Report</u>	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.6.3 is modified by replacing the references to the "facility" with the "plant" to reflect the power operation condition as follows: "The Radioactive Effluent Release Report covering operation of the ***plant*** ~~facility~~ in the previous year.…[for entirety of 5.6.3 text, see Attachment 1]…" This is an administrative change.

TS 5.6.5, <u>CORE OPERATING LIMITS REPORT (COLR)</u>	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.6.5 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. The COLR provides the required documentation and analytical methods used to determine the reactor core operating limits. The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. The COLR will be reestablished and controlled by this TS. Additionally, numerous reinstated TS reference the COLR for operating limits. Therefore, the requirements for this report are also reinstated.
TS 5.6.6, <u>Post Accident Monitoring Report</u>	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.6.6 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. Since TS 3.3.7, “Post Accident Monitoring (PAM) Instrumentation” is reinstated, the post-accident monitoring instrument reporting requirements from TS 3.3.7, Conditions B and G, will also reinstated, including references to TS 5.6.6. Therefore, this report is also reinstated.

TS 5.6.7, Containment Structural Integrity Surveillance Report	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.6.7 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 5.5.5, "Containment Structural Integrity Surveillance Program," is proposed to be reinstated in its entirety. It references this report. Therefore, this report is also reinstated.
TS 5.6.8, Steam Generator Tube Inspection Report	This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, TS 5.6.8 is reinstated as it existed in the previously approved TS prior to Amendment 272, to reflect the power operation condition of the plant. TS 5.5.8, "Steam Generator (SG) Program," which ensures that SG tube integrity is maintained, is proposed to be reinstated. It references this report. Therefore, this report is also reinstated.

3.2.3 Proposed Changes to RFOL Appendix B, Environmental Protection Plan

All changes to the EPP proposed in this LAR are made solely to more accurately reflect the PNP plant after it resumes power operation and to ensure the terminology used in the EPP is consistent with that used in the plant license. The proposed changes do not alter the obligations in the environmental area, including, as appropriate, requirements for reporting and keeping records of environmental data, and any conditions and monitoring requirement for the protection of the nonaquatic environment. As such, the changes to the EPP proposed by this LAR are administrative changes only.

The EPP is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, the EPP is reinstated as it existed in the previously approved RFOL prior to

Amendment 272, to reflect the power operation condition of the plant.

<table>
<tr><td><u>Current Table of Contents</u>

3.1 Facility Design and Operation

5.4 Facility Reporting Requirements</td><td><u>Proposed Table of Contents</u>

3.1 Plant ~~Facility~~ Design and Operation

5.4 Plant ~~Facility~~ Reporting Requirements</td></tr>
<tr><td colspan="2">Basis

To reflect PNP as an operating plant, the EPP is administratively revised to replace the word "Facility" with "Plant" to be consistent with terminology for the TS proposed in this LAR.</td></tr>
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<tr><td><u>Current Section 1.0</u>

1.0 "Objectives of the Environmental Protection Plan"

The Environmental Protection Plan (EPP) is to provide for protection of environmental values during handling and storage of spent fuel and maintenance of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is maintained in an environmentally acceptable manner, as established by the FES and other NRC environmental impact assessments. …

(3) Keep NRC informed of the environmental effects of handling and storage of spent fuel and maintenance of the facility and of actions taken to control those effects.</td><td><u>Proposed Section 1.0</u>

1.0 "Objectives of the Environmental Protection Plan"

The Environmental Protection Plan (EPP) is to provide for protection of environmental values during construction and operation ~~handling and storage of spent fuel and maintenance~~ of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the plant is operated ~~facility is maintained~~ in an environmentally acceptable manner, as established by the FES and other NRC environmental impact assessments. …

(3) Keep NRC informed of the environmental effects of ~~handling and storage of spent fuel and maintenance of the~~ facility construction and operation and of actions taken to control those effects.</td></tr>
<tr><td colspan="2">Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, the proposed administrative changes to Section 1.0 will replace a reference to "handling and storage of spent fuel and maintenance" with a reference to "construction and operation," a reference to "facility is maintained" with "plant is operated," and a reference to "handling and storage of spent fuel and maintenance of the facility" with</td></tr>
</table>

"facility construction and operation." These proposed administrative changes more accurately reflect the revised purpose of the plant in the operating condition.

Current Section 2.1	Proposed Section 2.1
2.1 Aquatic Issues …The need for aquatic monitoring programs to confirm that thermal mixing occurs as predicted, that chlorine releases are controlled within those discharge concentrations evaluated, and that effects on aquatic biota and water quality due to facility operation are no greater than predicted…	2.1 Aquatic Issues …The need for aquatic monitoring programs to confirm that thermal mixing occurs as predicted, that chlorine releases are controlled within those discharge concentrations evaluated, and that effects on aquatic biota and water quality due to ***plant*** ~~facility~~ operation are no greater than predicted…

Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, to reflect PNP as an operating plant, the EPP is administratively revised to replace the word "Facility" with "Plant" to be consistent with terminology for the TS proposed in this LAR.

Current Section 3.1	Proposed Section 3.1
3.1 Facility Design and Operation The licensee may make changes in facility design or operation or perform tests or experiments…Changes in facility design or operation or performance of tests or experiments… A proposed change, test, or experiment shall…(2) a significant change in effluents [in accordance with 10 CFR Part 51.5(b)(2)]…	3.1 ***Plant*** ~~Facility~~ Design and Operation The licensee may make changes in ***station*** ~~facility~~ design or operation or perform tests or experiments…Changes in ***plant*** ~~facility~~ design or operation or performance of tests or experiments… A proposed change, test, or experiment shall…(2) a significant change in effluents ***or power level*** [in accordance with 10 CFR Part 51.5(b)(2)]…

<table>
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Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, to reflect PNP as an operating plant, the EPP is administratively revised to replace the word "Facility" with "Plant" or “Station” to be consistent with terminology for the TS proposed in this LAR.

The proposed change to Section 3.1 to reinstate the reference to "power level [in accordance with 10 CFR Part 51.5(b)(2)]" reflects the operating condition of the plant. With the return to power operation of PNP, references to power level changes are now applicable. This proposed administrative change more accurately reflect the revised purpose of the plant.

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<u>Current Section 3.3</u>

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in Facility design or operation and…

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<u>Proposed Section 3.3</u>

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in ***plant*** ~~facility~~ design or operation and…

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Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, to reflect PNP as an operating plant, the EPP is administratively revised to replace the word "Facility" with "Plant" to be consistent with terminology for the TS proposed in this LAR.

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<u>Current Section 4.1</u>

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to the handling and storage of spent fuel and maintenance of the facility shall be recorded and…

</td><td>

<u>Proposed Section 4.1</u>

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to ***plant operation*** ~~the handling and storage of spent fuel and maintenance of the facility~~ shall be recorded and…

</td></tr>
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Basis

This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, the proposed change to Section 4.1 replaces "the handling and storage of spent fuel and maintenance of the facility" with "plant operation" to more accurately reflect the revised purpose of the facility in the operating condition. This proposed administrative change more accurately reflect the revised purpose of the plant and is consistent

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with the TS terminology proposed in this LAR.	
Current Section 5.2 5.2 Records Retention Records and logs relative to the environmental aspects of previous plant operation and the handling and storage of spent fuel and maintenance of the facility shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to the NRC on request. Records of modifications to facility structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the facility. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.	Proposed Section 5.2 5.2 Records Retention Records and logs relative to the environmental aspects of ~~previous~~ plant operation ~~and the handling and storage of spent fuel and maintenance of the facility~~ shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to the NRC on request. Records of modifications to ***plant*** ~~facility~~ structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the ***plant*** ~~facility~~. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.
Basis This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, the proposed changes to Section 5.2 removes the reference to "previous" plant operation and "the handling and storage of spent fuel and maintenance of the facility" to more accurately reflect the revised purpose of the facility in an operating condition. This proposed administrative change more accurately reflect the revised purpose of the plant in an operating condition and is consistent with the TS terminology proposed in this LAR. To reflect PNP as an operating plant, the EPP is administratively revised to replace the word "Facility" with "Plant" to be consistent with terminology for the TS proposed in this LAR.	
Current Section 5.4 5.4 Facility Reporting Requirements	Current Section 5.4 5.4 ***Plant*** ~~Facility~~ Reporting Requirements
Basis This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, to reflect PNP as an operating plant, the EPP is administratively revised to replace the word "Facility" with "Plant" to be consistent with terminology for the TS proposed in this LAR.	

Current Section 5.4.1	Proposed Section 5.4.1
5.4.1 Routine Reports …and an assessment of the observed impacts of the facility operation on the environment… (b) A list of all changes in facility design or operation, tests, and experiments…	5.4.1 Routine Reports …and an assessment of the observed impacts of the ***plant*** ~~facility~~ operation on the environment… (b) A list of all changes in ***station*** ~~facility~~ design or operation, tests, and experiments…
Basis	
This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, to reflect PNP as an operating plant, the EPP is administratively revised to replace the word "Facility" with "Plant" or “Station” to be consistent with terminology for the TS proposed in this LAR.	
Current Section 5.4.2	Proposed Section 5.4.2
5.4.2 Nonroutine Reports …The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and facility operating characteristics, (2)…	5.4.2 Nonroutine Reports …The report shall (1) describe, analyze, and evaluate the event, including extent and magnitude of the impact and ***plant*** ~~facility~~ operating characteristics, (2)…
Basis	
This TS is proposed for reinstatement in its entirety to that which was in effect prior to the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. Upon recission of the 10 CFR 50.82(a)(1) certifications, as conditioned by the exemption to 10 CFR 50.82(a)(2), Reference 3, to reflect PNP as an operating plant, the EPP is administratively revised to replace the word "Facility" with "Plant" to be consistent with terminology for the TS proposed in this LAR.	

3.2.4 Proposed Changes to the PNP Technical Specification Bases

Mark-ups of the TS Bases are provided in Attachment 3 for information only. Upon approval of this amendment, changes to the TS Bases will be incorporated in accordance with TS 5.5.12, *Technical Specifications (TS) Bases Control Program*.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements

<u>10 CFR 50.36, Technical Specifications</u>
In accordance with 10 CFR50.36, TS are required to include items in the following five

categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

This proposed amendment reinstates the portions of the previous PNP TS that are applicable to a power operation plant. It proposes changes to every section of the TS listed in 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The safety limits and limiting safety system settings are reinstated in Section 2.0 as described previously. The LCOs and SRs are reinstated to support plant power operation. The four criteria listed in 10 CFR 50.36(c)(2) are used to determine which structures, systems and components are required to be in the TS. As described in the previous sections, each TS in Sections 3.1 through 3.9 list the 10 CFR 50.36(c)(2) criterion that applies to a particular TS. The design features are revised to support the receipt and handling of new fuel for power operations. The administrative controls are modified and added to support the necessary programs and reports for an operating reactor. Therefore, the proposed PNP TS meet the criteria of 10 CFR 50.36 for an operating reactor.

10 CFR 50.36b, Environmental Conditions

10 CFR 50.36b states that TS may include conditions to protect the environment during operation and decommissioning. These conditions are set out in an attachment to the license and are derived from information contained in the environmental report or the supplement to the environmental report. Obligations in the environmental area, including, as appropriate, requirements for reporting and keeping records of environmental data, and any conditions and monitoring requirement for the protection of the nonaquatic environment are included. The proposed modifications to the EPP are administrative and do not change the environmental obligations required by 10 CFR 50.36b.

10 CFR 50.48(a) and (c), Fire Protection

These regulations establish the fire protection requirements for operating power reactors. 10 CFR 50.48(a) states: "Each holder of an operating license issued under this part…must have a fire protection plan that satisfies Criterion 3 of appendix A to this part." 10 CFR 50.48(c) provides the requirements for use of *National Fire Protection Association Standard NFPA 805* in creating a satisfactory fire protection plan. The reinstatement of License Condition 2.C.(3) addresses PNP compliance with these regulations.

10 CFR 50.51, Continuation of License

10 CFR 50.51 states, in part: "(a) Each license will be issued for a fixed period of time to be specified in the license…." PNP complies with this regulation by the reinstatement of the RFOL expiration date in the RFOL.

10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants

The General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The PNP design and licensing basis and its relationship to the General Design Criteria are described in the UFSAR, Revision 35 and other plant-specific licensing basis documents. UFSAR Revision 35 will be reinstated. This will include reinstatement of accident analyses and the safety reclassification of SSCs required to support the PNP POLB. Changes made to the UFSAR after Revision 35 will be evaluated for retention, to the extent appropriate for an operating plant.

10 CFR50, Appendix H, Reactor Vessel Material Surveillance Program Requirements
10 CFR 50 Appendix H requires that the design of the reactor vessel surveillance capsule program and withdrawal schedule must meet the requirements in the version of ASTM Standard Practice E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Code to which the reactor pressure vessel (RPV) was purchased. Reinstatement of License Condition 2.J addresses these requirements.

10 CFR 50, Appendix J, Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors
This section contains leakage test requirements, scheduled and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. The manner in which PNP complies with this regulation, including exemptions, is reinstated in License Condition 2.D.

4.2 Precedent

No nuclear power plant licensee to date has requested reauthorization of power operation after docketing the 10 CFR 50.82(a)(1) certifications and before reaching the renewed facility license expiration date. There have been instances in which a licensee submitted to the NRC, and then subsequently withdrew, a certification of an intent to cease operations under 10 CFR 50.82(a)(1)(i). In those cases, the licensee had not submitted on the docket the certification of permanent cessation of operation and permanent removal of fuel from the reactor vessel.

While current regulations do not specify a particular mechanism for reauthorizing operation of a nuclear power plant after both certifications are submitted on the docket and before operating license expiration, there is no statute or regulation prohibiting such action. Additionally, the NRC has considered the possibility of returning a plant to power operations as mentioned in Regulatory Guide 1.184, *Decommissioning of Nuclear Power Reactors* (Reference 13), and SECY-20-110, *Denial of Petition for Rulemaking on Criteria to Return Retired Power Reactors to Operations* (Reference 14). Thus, the NRC may address such requests under the existing regulatory framework—including granting exemptions, where needed—on a case-by-case basis. This proposed change to the RFOL, TS, and EPP supports the regulatory framework for reauthorization of power operations at PMP

Precedent for Removal of Technical Specification Table of Contents
Precedence for the removal of the Table of Contents from the Technical Specifications is found in approval of a request for Southern Nuclear Company in 2021. See Reference 15.

4.3 No Significant Hazards Consideration Determination

In accordance with 10 CFR 50.92, *Issuance of amendment,* Holtec Decommissioning International (HDI) has reviewed the proposed changes and concludes that the changes do not involve a significant hazards consideration since the proposed changes satisfy the criteria in 10 CFR 50.92(c). These criteria require that operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed license amendment would revise the Renewed Facility Operating License (RFOL), the Appendix A Permanently Defueled Technical Specifications (PDTS), and the Appendix B Environmental Protection Plan (EPP). The proposed changes are consistent with resumption of power operation of the reactor and emplacement and retention of fuel into the reactor vessel. The review of the proposed changes is based on the reinstatement of the plant operating licensing basis (POLB) as it was prior to the 10 CFR 50.82(a)(1) certifications. There are no physical changes to facility design proposed or required to support this amendment, and no changes proposed to the processes or procedures that were previously used during PNP power operations.

The proposed changes would revise certain requirements contained within the Palisades Nuclear Plant (PNP) RFOL, PDTS and EPP to add or revise license conditions or specifications that are necessary for power operation and revise or remove license conditions or specifications that would no longer be applicable. The proposed changes to the PNP RFOL and PDTS are in accordance with 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes to the PNP EPP are in accordance with 10 CFR 50.36b(b).

The discussion below addresses each 10 CFR 50.92(c) no significant hazards consideration criterion and demonstrates that the proposed amendment does not constitute a significant hazard.

1. <u>Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

   Response: No.

   The proposed changes to the PNP RFOL, PDTS and EPP do not impact the design basis function of plant structures, systems, or components (SSC). The proposed changes do not affect accident initiators or precursors, nor do they alter design assumptions that could increase the probability or consequences of previously evaluated accidents.

   Chapter 14 of the PNP Updated Final Safety Analysis Report (UFSAR) Revision 35 (ADAMS Accession No. ML21125A285) describes the postulated design basis accidents (DBA) and transient scenarios applicable to PNP during power operations. The UFSAR will be reinstated to reflect the docketed version (Revision 35) that was in effect prior to docketing the 10 CFR 50.82(a) certifications of permanent cessation of power operations and permanent removal of fuel at PNP. This will include restoration of the UFSAR Revision 35 which includes previously evaluated accident analyses and safety classification of SSCs to support power operations at PNP. The proposed changes to the PDTS simply revise and/or add license conditions or specifications applicable to the PNP POLB as previously evaluated in UFSAR Revision 35. The proposed changes do not involve physical changes to the facility or in the procedures governing operation of the plant that were in effect prior to 10 CFR 50.82(a)(1) certifications.

   The proposed addition / revision to TS definitions and rules of usage and application are those applicable to the reinstated PNP power operations technical specifications (TS) and have no impact on plant SSCs or the methods of operation of such SSCs.

The proposed reinstatement of PNP safety limits (SLs) and SL violations contain SLs that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity from the reactor core and the Primary Coolant System (PCS) pursuant to 10 CFR 50.36(c)(1). Since the proposed SLs are applicable to the power operations at PNP and provide protection of physical barriers to prevent uncontrolled radioactive release, they would not increase the probability or consequences of previously evaluated DBAs.

The reinstatement of TS Limiting Conditions for Operation (LCO) and Surveillance Requirements (SR) that are related to the operation of the nuclear reactor or to the prevention, diagnosis, or mitigation of reactor-related transients or accidents do not affect the applicable DBAs previously evaluated in the reinstated UFSAR. The safety functions involving core reactivity control, reactor heat removal, primary coolant system inventory control, and containment integrity are applicable at PNP as a power operation plant.

The proposed reinstatement of PNP design features contain features of the plant such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety pursuant to 10 CFR 50.36(c)(4). Since the proposed design features are applicable to the power operations at PNP and provide protection of important design features, they would not increase the probability or consequences of previously evaluated DBAs.

The addition and modification of provisions of the administrative controls of the PDTS and the non-radiological environmental protection requirements in the EPP do not affect any accidents applicable during power operation of the plant.

The probability of occurrence of previously evaluated accidents in the UFSAR is not increased since reinstatement of the previously approved licensing basis, including the RFOL, PDTS and EPP, is bounded by the reinstated analyses. Additionally, the proposed changes do not impact the function of plant structures, systems, or components.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes to the PNP RFOL, the PDTS, and the EPP have no impact on plant structures, systems or components. The proposed changes do not involve installation of new equipment or modification of existing equipment that could create the possibility of a new or different kind of accident. Hence, the proposed changes do not result in a change to the way the facility or equipment is operated in a manner which could cause a new or different kind of accident initiator to be created.

The addition of TS that are related only to the operation of the nuclear reactor or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents, cannot result in different or more adverse failure modes or accidents than previously evaluated because the plant will be operated within the previously approved POLB.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3. <u>Does the proposed amendment involve a significant reduction in a margin of safety?</u>

Response: No.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed amendment would modify the PNP RFOL and PDTS by adding the portions of the RFOL and TS that are credited in the accident analyses for the DBAs in the reinstated UFSAR. Postulated DBAs involving reactor operation are applicable because the plant will be in a power operation condition. These proposed changes impact operation of the facility and its response to transients or DBAs by reinstating requirements for equipment that is related to the operation of the nuclear reactor or to the prevention, diagnosis, or mitigation of reactor-related transients or accidents. The changes ensure that equipment required to respond to DBAs and transients described in the UFSAR remain capable of performing their safety function. No accident analyses or safety analyses acceptance criteria will be affected by the proposed changes.

Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

Based on the above, HDI concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL EVALUATION

This amendment request meets the eligibility criteria for categorical exclusion from environmental review set forth in 10 CFR 51.22, *Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review*, paragraph (c)(9). In support of this conclusion, as described in Reference 3, an independent environmental review of potentially new and significant information, and environmental issues not addressed in the October 2006 *Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Supplement 27, Regarding Palisades Nuclear Plant* was performed. The review concluded that the proposed licensing actions environmental impacts are consistent with the findings in the PNP RFOL Supplemental Environmental Impact Statement (NUREG 1427, Supplement 27), and hence the NRC staff recommendation to the Commission is applicable to this activity. The 10 CFR 51.22(c)(9) criteria are met as follows:

(i) The amendment involves no significant hazard consideration.

As described in Section 4.3 of this evaluation, the proposed amendment involves no significant hazards consideration. There are no changes to the design configuration or operation of the plant as constructed. There are no relaxations in the criteria used to establish safety limits or safety system settings or TS LCOs that were in effect prior to the 10 CFR 50.82(a)(1) certifications.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

There are no design configuration or operational changes proposed or required to support the reinstatement of the POLB that would change the type or amount of any effluents previously considered in the provisional, full-term, or renewed facility operating license environmental impact statements that considered power operations impacts through March 24, 2031. Reference 3 provides additional information. There are no expected changes in the types, characteristics, or quantities of effluents discharged to the environment associated with the proposed license amendment. The license amendment will not cause any materials or chemicals to be introduced into the plant that could affect the characteristics or types of effluents released offsite. Resumed power operations will be conducted under existing environmental permits. In addition, the method of operation of waste processing systems will not be affected by the proposed license amendment. The proposed license amendment will not result in changes to the design basis requirements of SSCs that function to limit or monitor the release of effluents. All the SSCs associated with limiting the release of effluents will continue to be able to perform the necessary functions.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

There are no design configuration or operational changes proposed or required to support reinstatement of the POLB that would change the cumulative public or occupational radiation exposure than previously considered in the provisional, full-term, or renewed facility operating license environmental impact statements that considered power operations impacts through March 24, 2031. Reference 3 provides additional information. Plant programs and processes to support an operating plant will be reinstated to ensure 10 CFR 20 limits are not exceeded for individual or cumulative occupational exposure. Since the proposed license amendment does not involve any physical change to the facility or in the procedures governing operation of the plant, the proposed license amendment does not involve a significant increase in individual or cumulative public or occupational radiation exposure.

Based on the above, HDI concludes that the proposed amendment meets the eligibility criteria for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 REFERENCES

1. Entergy Nuclear Operations, Inc. letter to U. S. Nuclear Regulatory Commission, "Supplement to Certification of Permanent Cessation of Power Operations," dated October 19, 2017 (ADAMS Accession No. ML17292A032)

2. Entergy Nuclear Operations, Inc. letter to U. S. Nuclear Regulatory Commission, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel." Dated June 13, 2022 (ADAMS Accession No. ML22164A067)

3. Holtec Decommissioning International letter to U. S. Nuclear Regulatory Commission, "Request for Exemption from Certain Termination of License Requirements of 10 CFR 50.82," dated September 28, 2023 (ADAMS Accession No. ML23271A140)

4. Entergy Nuclear Operations, Inc. letter to U. S. Nuclear Regulatory Commission, License Amendment Request to Revise Renewed Facility Operating License and Technical Specifications for Permanently Defueled Condition," dated June 1, 2021 (ADAMS Accession No. ML21152A108)

5. Entergy Nuclear Operations, Inc. letter to U.S. Nuclear Regulatory Commission, "License Amendment Request – Administrative Controls for a Permanently Defueled Condition," dated July 27, 2017 (ADAMS Accession No. ML17208A428)

6. Entergy Nuclear Operations, Inc. letter to U.S. Nuclear Regulatory Commission, "Final Safety Analysis Report Update – Revision 35," dated April 14, 2021, (ADAMS Accession No. ML21125A285)

7. U.S. Nuclear Regulatory Commission letter to Entergy Nuclear Operations, Inc., "Palisades Nuclear Plant and Big Rock Point Plant – Issuance of Amendment Nos. 129 and 273 re: Order Approving Transfer of Licenses and Conforming Administrative License Amendments," dated June 28, 2022 (ADAMS Accession No. ML22173A173)

8. U. S. Nuclear Regulatory Commission letter to Entergy Nuclear Operations, Inc., "Issuance of Amendment No. 272 re: Permanently Defueled Technical Specifications," dated May 13, 2022 (ADAMS Accession No. ML22039A198)

9. U. S. Nuclear Regulatory Commission letter to Entergy Nuclear Operations, Inc., "Palisades Plant – Issuance of Amendment Re: Alternative Radiological Source Term," dated September 28, 2007 (ADAMS Accession No. ML072470676)

10. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, dated July 2000.

11. Holtec Decommissioning International letter to U. S. Nuclear Regulatory Commission, "Application for Order Consenting to Transfer of Control of License and Approving Conforming License Amendments", dated December 6, 2023 (ADAMS Accession Nos. ML23340A161, ML23340A162)

12. U. S. Nuclear Regulatory Commission letter to Entergy Nuclear Operations, Inc., "Palisades Plant – Issuance of Amendment No. 271 Regarding Adoption of TSTF-425, Relocate Surveillance Frequencies to License Control – RITSTF Initiative 5B," dated December 30, 2019 (ADAMS Accession No. ML19317D855)

13. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.184, "Decommissioning of Nuclear Power Reactors," Revision 1, dated October 4, 2013 (ADAMS Accession No. ML13144A840)

14. U. S. Nuclear Regulatory Commission, SECY-20-110: Enclosure 1 – Federal Register Notice – Denial of Petition for Rulemaking on Criteria to Return Retired Nuclear Power Reactors to Operations (PRM 50-117; NRC 2019-0063), dated December 7, 2020 (ADAMS Accession No. ML20205L307)

15. U. S. Nuclear Regulatory Commission letter to Southern Nuclear Operating Co., Inc, "Issuance of Amendments Regarding Removal of Table of Contents from the Technical Specifications," dated September 29, 2021 (ADAMS Accession No. ML21232A149)

7.0 ATTACHMENTS

1. Proposed Changes (mark-up) to Palisades Plant Renewed Facility Operating License DPR-20, Appendix A Permanently Defueled Technical Specifications, and Appendix B Environmental Protection Plan Pages

2. Page Change Instructions and Retyped Pages for the Palisades Plant Renewed Facility Operating License DPR-20, Appendix A Technical Specifications, and Appendix B Environmental Protection Plan

3. Proposed Technical Specifications Bases Changes (for information only)

Enclosure Attachment 1 to

HDI PNP 2023-030

Proposed Changes (mark-up) to Palisades Plant

Renewed Facility Operating License DPR-20,

Appendix A Permanently Defueled Technical Specifications,

and

Appendix B Environmental Protection Plan Pages

Note, references to "HDI" are replaced by bracketed Palisades Energy, LLC, or Palisades Energy (e.g. [Palisades Energy]) to reflect the change in operating authority per license transfer application conforming amendments.

86 pages follow

HOLTEC PALISADES, LLC

~~HOLTEC DECOMMISSIONING INTERNATIONAL, LLC~~**[PALISADES ENERGY, LLC]**

DOCKET NO. 50-255

PALISADES NUCLEAR PLANT

RENEWED FACILITY OPERATING LICENSE

Renewed License No. DPR-20

1. The Nuclear Regulatory Commission (NRC or the Commission) having previously made the findings set forth in Operating License No. DPR-20, dated February 21, 1991, has now found that:

    A. The application for Renewed Operating License No. DPR-20 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

    B. [deleted];

    C. Actions have been identified and have been or will be taken with respect to: (1) managing the effects of aging on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1) during the period of extended operation, and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3 for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations;

    D. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

    E. There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

    F. Holtec Palisades, LLC (Holtec Palisades) is financially qualified and [Palisades Energy, LLC (Palisades Energy)]~~Holtec Decommissioning International, LLC~~ is financially and technically qualified to engage in the activities authorized by this renewed operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

G. Holtec Palisades and ~~HDI~~[Palisades Energy] have satisfied the applicable provisions of 10 CFR Part 140, ~~"~~“Financial Protection Requirements and Indemnity Agreements~~"~~” of the Commission’s regulations;

H. The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public;

I. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this renewed Facility Operating License No. DPR-20, subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 (formerly Appendix D to Part 50), of the Commission’s regulations and all applicable requirements have been satisfied; and

J. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this renewed operating license will be in accordance with 10 CFR Parts 30, 40, and 70.

2. Renewed Facility Operating License No. DPR-20 is hereby issued to Holtec Palisades and ~~HDI~~[Palisades Energy] as follows:

A. This renewed license applies to the Palisades Plant, a pressurized light water moderated and cooled reactor and electrical generating equipment (the facility). The facility is located in Van Buren County, Michigan, and is described in the Palisades Plant Updated Final Safety Analysis Report, as supplemented and amended, and in the Palisades Plant Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, “Licensing of Production and Utilization Facilities,” (a) Holtec Palisades to possess and use, and (b) ~~HDI~~[Palisades Energy] to possess, ~~and~~ use~~,~~ and operate, the facility as a utilization facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;

(2) ~~HDI~~[Palisades Energy], pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess and use source~~,~~ and special nuclear material ~~that was used~~ as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) ~~HDI~~[Palisades Energy], pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources ~~that were used~~ for reactor startup, ~~sealed sources that were used for~~ reactor instrumentation, ~~and are used in the calibration of~~ radiation monitoring equipment calibration, and ~~that were used as~~ fission detectors in amounts as

required;

(4) ~~HDI~~ [Palisades Energy], pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) ~~HDI~~[Palisades Energy], pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials ~~that were~~ as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act; to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) ~~[deleted]~~[Palisades Energy] is authorized to operate the facility at steady state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. XXX~~273~~, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. [Palisades Energy]~~HDI~~ shall ~~maintain~~ operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) ~~[deleted]~~ Fire Protection

[Palisades Energy] shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment requests dated December 12, 2012, November 1, 2017, November 1, 2018, and March 8, 2019, as supplemented by letters dated February 21, 2013, September 30, 2013, October 24, 2013, December 2, 2013, April 2, 2014, May 7, 2014, June 17, 2014, August 14, 2014, November 4, 2014, December 18, 2014, and January 24, 2018, and May 28, 2019, as approved in the safety evaluations dated February 27, 2015, February 27, 2018, and August 20, 2019. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than $1x10^{-7}$/year (yr) for CDF and less than $1x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) Other Changes that May Be Made Without Prior NRC Approval

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3 element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated February 27, 2015, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2, below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-2, "Plant Modifications Committed," of Entergy Nuclear Operations, Inc. (ENO) letter PNP 2019-028 dated May 28, 2019, to complete the transition to full compliance with 10 CFR 50.48(c) before the end of the refueling outage following the fourth full operating cycle after NRC approval. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Table S-3, "Implementation Items," of ENO letter PNP 2014-097 dated November 4, 2014, within six months after NRC approval, or six months after a refueling outage if in progress at the time of approval with the exception of Implementation Items 3 and 8 which will be completed once the related modifications are installed and validated in the PRA model.

(4) [deleted]

(5) ~~Movement of a fuel cask in or over the spent fuel pool is prohibited when irradiated fuel assemblies decayed less than 90 days are in the spent fuel pool.~~[deleted]

(6) Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

a. Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

b. Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

c. Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(7) [deleted]

(8) Amendment 257 authorizes the implementation of 10 CFR 50.61a in lieu of 10 CFR 50.61.~~[deleted]~~

D. The facility has been granted certain exemptions from Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors." This section contains leakage test requirements, scheduled and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted in a letter dated December 6, 1989.

These exemptions granted pursuant to 10 CFR 50.12, are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.~~[deleted]~~

E. ~~HDI~~[Palisades Energy] shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Palisades Nuclear Plant Physical Security Plan."

~~HDI~~[Palisades Energy] shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Palisades CSP was approved by License Amendment No. 243 as supplemented by changes approved by License Amendment Nos. 248, 253, 259, and 264.

F. [deleted]

G. Holtec Palisades and ~~HDI~~[Palisades Energy] shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. [deleted]

I. [deleted]

J. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal scheduled, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.~~[deleted]~~

K. This license is effective as of the date of issuance and shall expire at midnight March 24, 2031~~until the Commission notifies the licensee in writing that the license is terminated~~.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A – ~~Permanently Defueled~~ Technical Specifications

2. Appendix B – Environmental Protection Plan

Date of Issuance: January 17, 2007

<u>PALISADES PLANT</u>

<u>RENEWED FACILITY OPERATING LICENSE DPR-20</u>

<u>APPENDIX A</u>

~~PERMANENTLY DEFUELED~~
TECHNICAL SPECIFICATIONS

As Amended through Amendment No. ~~273~~

1.1 DEFINITIONS

1.0 USE AND APPLICATION

1.1 Definitions

--NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AVERAGE DISINTEGRATION ENERGY - Ē	Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the primary coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
AXIAL OFFSET (AO)	AO shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core (determined using the incore monitoring system).
AXIAL SHAPE INDEX (ASI)	ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core (determined using the excore monitoring system).
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training and retraining program required by Specification 5.3.2.

1.1 DEFINITIONS

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST.

Calibration of instrument channels with Resistance Temperature Detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

Whenever a RTD or thermocouple sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog and bistable channels - the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY;

1.1 DEFINITIONS

CHANNEL FUNCTIONAL TEST (continued)	b. Digital channels - the use of diagnostic programs to test digital hardware and the injection of simulated process data into the channel to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or control rods within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion,” 1989; (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

1.1 DEFINITIONS

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except Primary Coolant Pump seal water leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known not to interfere with the operation of leakage detection systems and not to be pressure boundary LEAKAGE; and

3. Primary Coolant System (PCS) LEAKAGE through a Steam Generator to the Secondary System (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

All LEAKAGE (except Primary Coolant Pump seal leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in a PCS component body, pipe wall, or vessel wall.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average primary coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

NON-CERTIFIED OPERATOR

A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 5.3.1.

1.1 DEFINITIONS

OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 13, Initial Tests and Operation, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT (T_q)	T_q shall be the maximum positive ratio of the power generated in any quadrant minus the average quadrant power, to the average quadrant power.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the primary coolant of 2565.4 MWt.
REFUELING BORON CONCENTRATION	REFUELING BORON CONCENTRATION shall be a Primary Coolant System boron concentration of $\geq$ 1720 ppm and sufficient to assure the reactor is subcritical by $\geq$ 5% $\Delta\rho$ with all control rods withdrawn.

1.1 DEFINITIONS

SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length control rods (shutdown and regulating) are fully inserted except for the single rod of highest reactivity worth, which is assumed to be fully withdrawn. However, with all full length control rods verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any full length control rods not capable of being fully inserted, the reactivity worth of these rods must be accounted for in the determination of SDM; and b. There is no change in part length rod position
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the primary coolant.
TOTAL RADIAL PEAKING FACTOR (F_R^T)	F_R^T shall be the maximum ratio of the individual fuel pin power to the core average pin power integrated over the total core height, including tilt.

1.1 DEFINITIONS

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE PRIMARY COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 300
4	Hot Shutdown[b]	< 0.99	NA	$300 > T_{ave} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear~~s~~ in TS ~~is~~ are AND and OR. The physical arrangement of ~~this~~ these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several ~~Levels~~ levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

1.2 Logical Connectors

EXAMPLES

The following examples illustrate~~s~~ the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 ~~Suspend~~ Verify . . . AND A.2 ~~Initiate~~ Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.3 Completion Times

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe ~~storage and handling of spent nuclear fuel~~operation of the plant. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the ~~facility~~ plant is in a MODE or specified condition stated in the Applicability of the LCO. Unless otherwise specified, ~~The~~ the Completion Time begins when a ~~Certified Fuel Handler (CFH)~~senior licensed operator on the operating shift crew with responsibility for plant operations makes the determination that an LCO is not met and an ACTIONS Condition is entered. The "otherwise specified" exceptions are varied, such as a Required Action Note or Surveillance Requirement Note that provides an alternative time to perform specific tasks, such as testing, without starting the Completion Time. While utilizing the Note, should a Condition be applicable for any reason not addressed by the Note, the Completion Time begins. Should the time allowance in the Note be exceeded, the Completion Time begins at that point. The exceptions may also be incorporated into the Completion Time. For example, LCO 3.8.1, "AC Sources - Operating," Required Action B.2, requires declaring required feature(s) supported by an inoperable diesel generator, inoperable when the redundant required feature(s) are inoperable. The Completion Time states, "4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)." In this case the Completion Time does not begin until the conditions in the Completion Time are satisfied. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the plant is not within the LCO Applicability.

1.3 Completion Times

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the discovery of the situation that required entry into the Condition, unless otherwise specified.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition, unless otherwise specified.

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the

1.3 Completion Times

Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

EXAMPLES

The following examples illustrate~~s~~ the use of Completion Times with different types of Conditions and changing Conditions~~Required Actions~~.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A~~B. Required Action and associated Completion Time not met~~Spent Fuel Pool boron concentration not within limit~~.	~~A~~B.1 Be in MODE 3~~Suspend movement of fuel assemblies in the Spent Fuel Pool~~.	~~Immediately~~6 hours
	AND	
	~~A~~B.2 Be in MODE 5.~~Initiate action to restore Spent Fuel Pool boron concentration to within limit~~	~~Immediately~~36 hours

Condition ~~A~~ B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition ~~A~~ B is entered.

The Required Actions of Condition ~~A~~ B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

1.3 Completion Times

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.~~immediately suspend movement of fuel assemblies in the Spent Fuel Pool and initiate action to restore Spent Fuel Pool boron within limit.~~

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B.

EXAMPLES

EXAMPLE 1.3-2 (continued)

The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	12 hours 12 hours

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 30 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-5

ACTIONS

---NOTE---
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered.

1.3 Completion Times

EXAMPLES EXAMPLE 1.3-7 (continued)

The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE COMPLETION TIME When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" and "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

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1.4 Frequency

DESCRIPTION
(continued)

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.~~illustrate the type of Frequency statements that appears in the Technical Specifications (TS).~~

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK~~Verify level is within limit~~.	~~7 days~~12 hours

Example 1.4-1 contains ~~one~~ the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval ~~of 7 days~~(12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as ~~7 days~~12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the ~~facility~~ plant is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the ~~facility~~ plant is not in a Mode or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits. ~~...~~	Once within 12 hours after ≥ 25% RTP~~Prior to storing a fuel assembly...~~ AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is~~illustrates~~ a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE---------------------------- Not required to be performed until 12 hours after ≥ 25% RTP. --- Perform channel adjustment.	7 days

The interval continues, whether or not the plant operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." The interval continues, whether or not the plant operation is < 25% RTP between performances. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the plant reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------------NOTE--------------------------- Only required to be met in MODE 1. --- Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the plant is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an “otherwise stated” exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the plant was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------------NOTE--------------------------- Only required to be performed in MODE 1. --	
Perform complete cycle of the valve.	7 days

The interval continues, whether or not the plant operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the plant reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------------NOTE--------------------------- Not required to be met in MODE 3. --- Verify parameter is within limits.	24 hours

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the plant is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the plant was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

2.0 SAFETY LIMITS (SLs)

2.~~0~~1 ~~(Deleted)~~SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained at or above the following DNB correlation safety limits:

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

2.1.1.2 In MODES 1 and 2, the peak Linear Heat Rate (LHR) (adjusted for fuel rod dynamics) shall be maintained at $\leq$ 21.0 kW/ft.

2.1.2 Primary Coolant System (PCS) Pressure SL

In MODES 1, 2, 3, 4, 5, and 6, the PCS pressure shall be maintained at $\leq$ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, 5, or 6, restore compliance within 5 minutes.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the plant shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the plant, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 31 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate (exceptions to this Specification are stated in the individual Specifications); or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the plant.

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

3.0 LCO APPLICABILITY

LCO 3.0.7 Special Test Exception (STE) LCOs in each applicable LCO section allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

LCO 3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

3.0 LCO APPLICABILITY

LCO 3.0.9 (continued)

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

3.0 SR APPLICABILITY

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the plant.

3.1 REACTIVITY CONTROL SYSTEMS

Insert LCO 3.1.1	SHUTDOWN MARGIN (SDM)
Insert LCO 3.1.2	Reactivity Balance
Insert LCO 3.1.3	Moderator Temperature Coefficient (MTC)
Insert LCO 3.1.4	Control Rod Alignment
Insert LCO 3.1.5	Shutdown and Part-Length Control Rod Group Insertion Limits
Insert LCO 3.1.6	Regulating Rod Group Position Limits
Insert LCO 3.1.7	Special Test Exceptions (STE)

3.2 POWER DISTRIBUTION LIMITS

Insert LCO 3.2.1	Linear Heat Rate (LHR)
Insert LCO 3.2.2	TOTAL RADIAL PEAKING FACTOR (F_R^T)
Insert LCO 3.2.3	QUADRANT POWER TILT (Tq)
Insert LCO 3.2.4	AXIAL SHAPE INDEX (ASI)

3.3 INSTRUMENTATION

Insert LCO 3.3.1	Reactor Protective System (RPS) Instrumentation
Insert LCO 3.3.2	Reactor Protective System (RPS) Logic and Trip Initiation
Insert LCO 3.3.3	Engineered Safety Features (ESF) Instrumentation
Insert LCO 3.3.4	Engineered Safety Features (ESF) Logic and Manual Initiation
Insert LCO 3.3.5	Diesel Generator (DG) - Undervoltage Start (UV Start)
Insert LCO 3.3.6	Refueling Containment High Radiation (CHR) Instrumentation
Insert LCO 3.3.7	Post Accident Monitoring (PAM) Instrumentation
Insert LCO 3.3.8	Alternate Shutdown System
Insert LCO 3.3.9	Neutron Flux Monitoring Channels
Insert LCO 3.3.10	Engineered Safeguards Room Ventilation (ESRV) Instrumentation

3.4 PRIMARY COOLANT SYSTEM (PCS)

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

Insert LCO 3.4.2 PCS Minimum Temperature for Criticality

Insert LCO 3.4.3 PCS Pressure and Temperature (P/T) Limits

Insert LCO 3.4.4 PCS Loops - MODES 1 and 2

Insert LCO 3.4.5 PCS Loops - MODE 3

Insert LCO 3.4.6 PCS Loops - MODE 4

Insert LCO 3.4.7 PCS Loops - MODE 5, Loops Filled

Insert LCO 3.4.8 PCS Loops - MODE 5, Loops Not Filled

Insert LCO 3.4.9 Pressurizer

Insert LCO 3.4.10 Pressurizer Safety Valves

Insert LCO 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

Insert LCO 3.4.12 Low Temperature Overpressure Protection (LTOP) System

Insert LCO 3.4.13 PCS Operational LEAKAGE

Insert LCO 3.4.14 PCS Pressure Isolation Valve (PIV) Leakage

Insert LCO 3.4.15 PCS Leakage Detection Instrumentation

Insert LCO 3.4.16 PCS Specific Activity

Insert LCO 3.4.17 Steam Generator (SG) Tube Integrity

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

Insert LCO 3.5.1 Safety Injection Tanks (SITs)

Insert LCO 3.5.2 ECCS - Operating

Insert LCO 3.5.3 ECCS - Shutdown

Insert LCO 3.5.4 Safety Injection Refueling Water Tank (SIRWT)

Insert LCO 3.5.5 Containment Sump Buffering Agent and Weight Requirements

3.6 CONTAINMENT SYSTEMS

Insert LCO 3.6.1	Containment
Insert LCO 3.6.2	Containment Air Locks
Insert LCO 3.6.3	Containment Isolation Valves
Insert LCO 3.6.4	Containment Pressure
Insert LCO 3.6.5	Containment Air Temperature
Insert LCO 3.6.6	Containment Cooling Systems

3.7 PLANT SYSTEMS

Insert LCO 3.7.1	Main Steam Safety Valves (MSSVs)
Insert LCO 3.7.2	Main Steam Isolation Valves (MSIVs)
Insert LCO 3.7.3	Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves
Insert LCO 3.7.4	Atmospheric Dump Valves (ADVs)
Insert LCO 3.7.5	Auxiliary Feedwater (AFW) System
Insert LCO 3.7.6	Condensate Storage and Supply
Insert LCO 3.7.7	Component Cooling Water (CCW) System
Insert LCO 3.7.8	Service Water System (SWS)
Insert LCO 3.7.9	Ultimate Heat Sink (UHS)
Insert LCO 3.7.10	Control Room Ventilation (CRV) Filtration
Insert LCO 3.7.11	Control Room Ventilation (CRV) Cooling
Insert LCO 3.7.12	Fuel Handling Area Ventilation System
Insert LCO 3.7.13	Engineered Safeguards Room Ventilation (ESRV) Dampers

3.7 ~~FACILITY~~ PLANT SYSTEMS

3.7.14 Spent Fuel Pool (SFP) Water Level

LCO 3.7.14 The SFP water level shall be ≥ 647 ft elevation.

--NOTE---
SFP level may be below the 647 ft elevation to support fuel cask movement, if the displacement of water by the fuel cask when submerged in the SFP, would raise SFP level to ≥ 647 ft elevation.

APPLICABILITY: During movement of irradiated fuel assemblies in the SFP,
During movement of a fuel cask in or over the SFP.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies in SFP. AND A.2 Suspend movement of fuel cask in or over the SFP.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the SFP water level is ≥ 647 ft elevation.	In accordance with the Surveillance Frequency Control Program~~7 days~~

3.7 ~~FACILITY~~ PLANT SYSTEMS

3.7.15 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.15 The SFP boron concentration shall be ≥ 1720 ppm.

APPLICABILITY: When fuel assemblies are stored in the Spent Fuel Pool.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the SFP. AND A.2 Initiate action to restore SFP boron concentration to within limit.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.15.1 Verify the SFP boron concentration is within limit.	In accordance with the Surveillance Frequency Control Program~~7 days~~

3.7 ~~FACILITY~~ PLANT SYSTEMS

3.7.16 Spent Fuel Pool Storage

LCO 3.7.16 Storage in the spent fuel pool shall be as follows:

a. Each fuel assembly and non-fissile bearing component stored in a Region I Carborundum equipped storage rack shall be within the limitations in Specification 4.3.1.1 and, as applicable, within the requirements of the maximum nominal planar average U-235 enrichment and burnup of Tables 3.7.16-2, 3.7.16-3, 3.7.16-4 or 3.7.16-5,

b. Fuel assemblies in a Region I Metamic equipped storage rack shall be within the limitations in Specification 4.3.1.2, and

c. The combination of maximum nominal planar average U-235 enrichment, burnup, and decay time of each fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly or non-fissile bearing component is stored in the spent fuel pool or the north tilt pit.

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to restore the noncomplying fuel assembly or non-fissile bearing component within requirements.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means each fuel assembly or non-fissile bearing component meets fuel storage requirements.	Prior to storing the fuel assembly or non-fissile bearing component in the spent fuel pool

3.7 PLANT SYSTEMS

Insert LCO 3.7.17 Secondary Specific Activity

3.8 ELECTRICAL POWER SYSTEMS

Insert LCO 3.8.1 AC Sources - Operating

Insert LCO 3.8.2 AC Sources - Shutdown

Insert LCO 3.8.3 Diesel Fuel, Lube Oil, and Starting Air

Insert LCO 3.8.4 DC Sources - Operating

Insert LCO 3.8.5 DC Sources - Shutdown

Insert LCO 3.8.6 Battery Cell Parameters

Insert LCO 3.8.7 Inverters - Operating

Insert LCO 3.8.8 Inverters - Shutdown

Insert LCO 3.8.9 Distribution Systems - Operating

Insert LCO 3.8.10 Distribution Systems - Shutdown

3.9 REFUELING OPERATIONS

Insert LCO 3.9.1 Boron Concentration

Insert LCO 3.9.2 Nuclear Instrumentation

Insert LCO 3.9.3 Containment Penetrations

Insert LCO 3.9.4 Shutdown Cooling (SOC) and Coolant Circulation - High Water Level

Insert LCO 3.9.5 Shutdown Cooling (SOC) and Coolant Circulation - Low Water Level

Insert LCO 3.9.6 Refueling Cavity Water Level

4.0 DESIGN FEATURES

4.1 Site Location

The Palisades Nuclear Plant is located on property owned by Entergy Nuclear Palisades, LLC on the eastern shore of Lake Michigan approximately four and one-half miles south of the southern city limits of South Haven, Michigan. The minimum distance to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 677 meters.

4.2 ~~(Deleted)~~ Reactor Core

4.2.1 Fuel Assemblies

The reactor core shall contain 204 fuel assemblies. Each assembly shall consist of a matrix of zircaloy-4 or M5 clad fuel rods with an initial composition of depleted, natural, or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution. Poison may be placed in the fuel bundles for long-term reactivity control.

4.2.2 Control Rod Assemblies

The reactor core shall contain 45 control rods. Four of these control rods may consist of part-length absorbers. The control material shall be silver-indium-cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The Region I (See Figure B 3.7.16-1) Carborundum equipped fuel storage racks incorporating Regions 1A, 1B, 1C, 1D, and 1E are designed and shall be maintained with:

a. ~~Irradiated~~ New or irradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.54 weight percent;

4.3.1 Criticality (continued)

3. Control blades may be stored in both fueled and unfueled locations in Regions 1D and 1E, with no limitation on the number.

4.3.1.2 The Region I (See Figure B 3.7.16-1) Metamic equipped fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

d. A nominal 10.25 inch center to center distance between fuel assemblies;

e. New or ~~Irradiated~~ irradiated fuel assemblies;

f. Two empty rows of storage locations shall exist between the fuel assemblies in a Carborundum equipped rack and the fuel assemblies in an adjacent Metamic equipped rack; and

g. A minimum Metamic B^{10} areal density of 0.02944 g/cm^2.

4.3.1.3 The Region II fuel storage racks (See Figure B 3.7.16-1) are designed and shall be maintained with:

a. Fuel assemblies having maximum nominal planar average U-235 enrichment of 4.60 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR.

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowance for uncertainties as described in Section 9.11 of the FSAR.

d. A nominal 9.17 inch center to center distance between fuel assemblies; and

e. New or ~~Irradiated~~ irradiated fuel assemblies which meet the maximum nominal planar average U-235 enrichment, burnup, and decay time requirements of Table 3.7.16-1

4.3 Fuel Storage

4.3.1 Criticality (continued)

4.3.1.4 ~~(Deleted)~~The new fuel storage racks are designed and shall be maintained with:

a. Twenty four unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1, or

Thirty six unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.05 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1;

b. Keff ≤ 0.95 when flooded with either full density or low density (optimum moderation) water including allowances for uncertainties as described in Section 9.11 of the FSAR.

c. The pitch of the new fuel storage rack lattice being ≥ 9.375 inches and every other position in the lattice being permanently occupied by an 8" x 8" structural steel or core plugs, resulting in a nominal 13.26 inch center to center distance between fuel assemblies placed in alternating storage locations.

4.3.2 Drainage

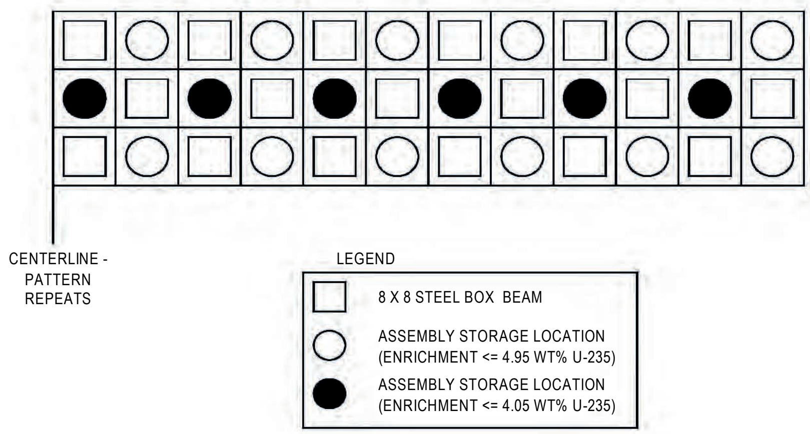
The spent fuel storage pool cooling system suction and discharge piping is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 644 ft 5 inches.

4.3.3 Capacity

The spent fuel storage pool and north tilt pit are designed and shall be maintained with a storage capacity limited to no more than 892 fuel assemblies.

4.3 Fuel Storage

INSERT FIGURE 4.3-1



Note: If any assemblies containing fuel enrichments greater than 4.05% U-235 are stored in the New Fuel Storage Rack, the center row must remain empty.

Figure 4.3-1 (page 1 of 1)
New Fuel Storage Rack Arrangement

5.5 Programs and Manuals

5.5.2 ~~(Deleted)~~Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage to the engineered safeguards rooms, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident, to as low as practical. The systems include the Containment Spray System, the Safety Injection System, the Shutdown Cooling System, and the containment sump suction piping. This program shall include the following:

a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

c. The portion of the shutdown cooling system that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 255 psig.

d. Piping from valves CV-3029 and CV-3030 to the discharge of the safety injection pumps and containment spray pumps shall be hydrostatically tested at no less than 100 psig.

e. The maximum allowable leakage from the recirculation heat removal systems' components (which include valve stems, flanges and pump seals) shall not exceed 0.2 gallon per minute under the normal hydrostatic head from the SIRW tank.

5.5.3 (Deleted)

5.5.4 Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the Offsite Dose Calculation Manual (ODCM), (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

5.5 Programs and Manuals

5.5.5 ~~(Deleted)~~ Containment Structural Integrity Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Containment Structural Integrity Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE and IWL.

If, as a result of a tendon inspection, corrective retensioning of five percent (8) or more of the total number of dome tendons is necessary to restore their liftoff forces to within the limits, a dome delamination inspection shall be performed within 90 days following such corrective retensioning. The results of this inspection shall be reported to the NRC in accordance with Specification 5.6.7, "Containment Structural Integrity Surveillance Report."

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Containment Structural Integrity Surveillance Program inspection frequencies.

5.5.6 ~~(Deleted)~~ Primary Coolant Pump Flywheel Surveillance Program

a. Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the upper flywheels each 10 years.

b. The provisions of SR 3.0.2 are not applicable to the Flywheel Testing Program

5.5.7 (Deleted)

5.5.8 ~~(Deleted)~~#Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.3 gpm.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, “PCS Operational LEAKAGE.”

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged. The following alternative repair criteria shall be applied as an alternate to the 40% depth based criteria:

1. Tubes found by inservice inspection to contain service-induced flaws within 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, shall be plugged. Flaws located below this elevation may remain in service.

2. Tubes found by inservice inspection to contain service-induced flaws within 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, shall be plugged. Flaws located below this elevation may remain in service.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

5.5.8 Steam Generator (SG) Program

d. Provisions for SG tube inspections. (continued)

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

3. If crack indications are found in any SG tube from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

4. When the alternate repair criteria of TS 5.5.8c.1 are implemented, inspect 100% of the inservice tubes to the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 5.5.8c.1 every 24 effective full-power months, or one refueling outage, whichever is less.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

5.5.9 ~~(Deleted)~~ Secondary Water Chemistry Program

A program shall be established, implemented and maintained for monitoring of secondary water chemistry to inhibit steam generator tube degradation and shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical variables,

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

d. Procedures for the recording and management of data,

e. Procedures defining corrective actions for all off-control point chemistry conditions, and

f. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

5.5.10 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of Control Room Ventilation (CRV) and Fuel Handling Area Ventilation (FHAV) systems at the frequencies specified in Regulatory Guide 1.52, Revision 2 (RG 1.52), and in accordance with RG 1.52 and ASME N510-1989, at the system flowrates and tolerances specified below*:

a. Demonstrate for each of the ventilation systems that an inplace test of the High Efficiency Particulate Air (HEPA) filters shows a penetration and system bypass < 0.05% for the CRV system and < 1.00% for the FHAV system when tested in accordance with RG 1.52 and ASME N510-1989: ~~(Deleted)~~

Ventilation System	Flowrate (CFM)
FHAV (single fan operation)	7300 ± 20%
FHAV (dual fan operation)	10,000 ± 20%
CRV	3,200 +10% -5%

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (Continued

b. Demonstrate for each of the ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% for the CRV system and < 1.00% for the FHAV system when tested in accordance with RG 1.52 and ASME N510-1989.

Ventilation System	Flowrate (CFM)
FHAV (dual fan operation)	10,000 ± 20%
CRV	3200 +10% -5%

c. Demonstrate for each of the ventilation systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in RG 1.52 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and equal to the relative humidity specified as follows:

Ventilation System	Penetration	Relative Humidity
FHAV	6.00%	95%
CRV	0.157%	70%

d. For each of the ventilation systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Delta P (In H_20)	Flowrate (CFM)
FHAV (dual fan operation)	6.0	10,000 ± 20%
CRV	8.0	3200 +10% -5%

e. Demonstrate that the heaters for the CRV system dissipates the following specified value ± 20% when tested in accordance with ASME N510-1989:

Ventilation System	Wattage
CRV	15 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Ventilation Filter Testing Program frequencies.

* Should the 720-hour limitation on charcoal adsorber operation occur during a plant operation requiring the use of the charcoal adsorber - such as refueling - testing may be delayed until the completion of the plant operation or up to 1,500 hours of filter operation; whichever occurs first.

5.5 Programs and Manuals

5.5.11 ~~(Deleted)~~ Fuel Oil Testing Program

A fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling requirements, testing requirements, and acceptance criteria, based on the diesel manufacturer's specifications and applicable ASTM Standards. The program shall establish the following:

a. Acceptability of new fuel oil prior to addition to the Fuel Oil Storage Tank, and acceptability of fuel oil stored in the Fuel Oil Storage Tank, by determining that the fuel oil has the following properties within limits:

   1. API gravity or an absolute specific gravity,

   2. Kinematic viscosity, and

   3. Water and sediment content.

b. Other properties of fuel oil stored in the Fuel Oil Storage Tank, specified by the diesel manufacturers or specified for grade 2D fuel oil in ASTM D 975, are within limits.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Fuel Oil Testing Program.

5.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5 Programs and Manuals

5.5.12 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.12.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e)

5.5.13 ~~(Deleted)~~ Safety Functions Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.5.5.13 Safety Functions Determination Program (SFDP) (Continued)

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

5.5.13 Safety Functions Determination Program (SFDP) (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 ~~(Deleted)~~ Containment Leak Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008, with the following exceptions:

1. Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

   Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at $\geq$10 psig instead.

3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 54.2 psig. The containment design pressure is 55 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

5.5 Programs and Manuals

5.5.14 Containment Leak Rate Testing Program (Continued)

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage is $\leq 1.0\ L_a$ when tested at $\geq P_a$ and combined with all penetrations and valves subjected to Type B and C tests. However, during the first unit startup following testing performed in accordance with this program, the leakage rate acceptance criteria is $< 0.6\ L_a$ when combined with all penetrations and valves subjected to Type B and C tests.

b) For each Personnel Air Lock door, leakage is $\leq 0.023\ L_a$ when pressurized to ≥ 10 psig.

c) For each Emergency Escape Air Lock door, a seal contact check , consisting of a verification of continuous contact between the seals and the sealing surfaces, is acceptable.

e. “Containment OPERABILITY” is equivalent to "Containment Integrity" for the purposes of the testing requirements.

f. The provisions of SR 3.0.3 are applicable to the Containment Leak Rate Testing Program requirements.

g. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5 Programs and Manuals

5.5.16 ~~(Deleted)~~ Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation (CRV) Filtration, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRV Filtration, operating at the flow rate required by the Ventilation Filter Testing Program, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5 Programs and Manuals

5.5.17 ~~(Deleted)~~ Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 (Deleted)

5.6.2 Radiological Environmental Operating Report

The Radiological Environmental Operating Report covering the operation of the ~~facility~~ plant during the previous calendar year shall be submitted before May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering operation of the ~~facility~~ plant in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the ~~facility~~plant. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual and Process Control Program, and shall be in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 (Deleted)

5.6.5 ~~(Deleted)~~ CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1 Shutdown Margin
3.1.6 Regulating Rod Group Position Limits
3.2.1 Linear Heat Rate Limits
3.2.2 Radial Peaking Factor Limits
3.2.4 ASI Limits
3.4.1 DNB Limits

5.6.5 COLR (Continued)

b. The analytical methods used to determine the core operating limits shall be those approved by the NRC, specifically those described in the latest approved revision of the following documents:

1. EMF-96-029(P)(A) Volumes 1 and 2, “Reactor Analysis System for PWRs,” Siemens Power Corporation. (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

2. ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation. (Bases report not approved) (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

4. EMF-84-093(P)(A), “Steam Line Break Methodology for PWRs, “Siemens Power Corporation. (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

5. XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company. (Bases document not approved) (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

6. EMF-2310 (P)(A), Revision 0, Framatome ANP, Inc., May 2001, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors.” (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

7. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, & 3.2.2)

8. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

9. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation. (LCOs 3.2.1, 3.2.2, & 3.2.4)

5.6.5 COLR (Continued)

10. XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

11. XN-NF-82-06(P)(A) and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

12. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWD/MTU," Advanced Nuclear Fuels Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

13. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

14. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

15. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation. (LCOs 3.1.6, 3.2.1, & 3.2.2)

16. ANF-87-150 Volume 2, "Palisades Modified Reactor Protection System Report: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation. [Approved for use in the Palisades design during the NRC review of license Amendment 118, November 15, 1988] (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.4.1)

17. EMF-1961(P)(A), Revision 0, Siemens Power Corporation, July 2000, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors." (LCOs 3.1.6, 3.2.1, 3.2.2, 3.2.4, & 3.4.1)

18. EMF-2328 (P)(A), Revision 0, Framatome ANP, Inc., March 2001, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." (LCOs 3.1.6, 3.2.1, & 3.2.2)

19. BAW-2489P, "Revised Fuel Assembly Growth Correlation for Palisades." (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

5.6.5 COLR (Continued)

20. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors." (LCOs 3.1.6, 3.2.1, & 3.2.2)

21. BAW-10240(P)-A, "Incorporation of M5 Properties in Framatome ANP Approved Methods." (LCOs 3.1.6, 3.2.1, 3.2.2, 3.2.4, & 3.4.1)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC.

5.6.6 ~~(Deleted)~~ Post Accident Monitoring Report

When a report is required by LCO 3.3.7, "Post Accident Monitoring Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to OPERABLE status.

5.6.7 ~~(Deleted)~~ Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Dome Delamination tests within 90 days after completion of the tests.

5.6.8 ~~(Deleted)~~ Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism, 5.6.8 Steam Generator Tube Inspection Report (Continued)

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

i The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

PALISADES PLANT

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

TABLE OF CONTENTS

Section Page

1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of environmental values during ~~handling and storage of spent fuel~~construction and ~~maintenance~~ operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the ~~facility is maintained~~plant is operated in an environmentally acceptable manner, as established by the FES and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of ~~handling and storage of spent fuel and maintenance of the~~ facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES which relate to water quality matters are regulated by way of the licensee's NPDES permit.

2.0 Environmental Protection Issues

In the final addendum to the FES-OL dated February 1978 the staff considered the environmental impacts associated with the operation of the Palisades Plant. Certain environmental issues were identified which required study or license conditions to resolve environmental concerns and to assure adequate protection of the environment.

2.1 Aquatic Issues

Specific aquatic issues raised by the staff in the FES-OL were:

The need for aquatic monitoring programs to confirm that thermal mixing occurs as predicted, that chlorine releases are controlled within those discharge concentrations evaluated, and that effects on aquatic biota and water quality due to ~~facility~~ plant operation are no greater than predicted.

Aquatic issues are addressed by the effluent limitations, and monitoring requirements are contained in the effective NPDES permit issued by the State of Michigan, Department of Natural Resources. The NRC will rely on this agency for regulation of matters involving water quality and aquatic biota.

2.2 Terrestrial Issues

1. Potential impacts on the terrestrial environment associated with drift from the mechanical draft cooling towers. (FES-OL addendum Section 6.3)

3.0 Consistency Requirements

3.1 ~~Facility~~ Plant Design and Operation

The licensee may make changes in ~~facility~~ station design or operation or perform tests or experiments affecting the environment provided such changes, tests or experiments do not involve an unreviewed environmental question, and do not involve a change in the Environmental Protection Plan. Changes in ~~facility~~ plant design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this section.

Before engaging in additional construction or operational activities which may affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation. When such activity involves a change in the Environmental Protection Plan, such activity and change to the Environmental Protection Plan may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the final environmental statement (FES) as modified by staff's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level [in accordance with 10 CFR Part 51.5(b)(2)] or (3) a matter not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in ~~facility~~ plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, or local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to ~~the handling and storage of spent fuel and maintenance of the facility~~plant operation shall be recorded and promptly reported to the NRC within 24 hours by telephone, telegraph, or facsimile transmissions followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events, onsite plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, fish kills, increase in nuisance organisms or conditions and unanticipated or emergency discharge of waste water or chemical substances.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Meteorological Monitoring

A meteorological monitoring program shall be conducted in the vicinity of the plant site for at least two years after conversion to cooling towers to document effects of cooling tower operation on meteorological variables. Data on the following meteorological variables shall be obtained from the station network shown in Figure 4.2.1: precipitation, temperature, humidity, solar radiation, downcoming radiation, visibility, wind direction and wind speed. In addition, studies shall be conducted for at least two years to measure affects of cooling tower drift on vegetation by associated salt deposition, icing or other causes.

5.0 Administrative Procedures

5.1 Review and Audit

The licensee shall provide for review and audit of compliance with the Environmental Protection Plan. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of ~~previous~~ plant operation ~~and the handling and storage of spent fuel and maintenance of the facility~~ shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to ~~facility~~ plant structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the ~~facility~~plant. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.3 Changes in Environmental Protection Plan

Request for change in the Environmental Protection Plan shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the Environmental Protection Plan.

5.4 ~~Facility~~ Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following issuance of the operating license. The period of the first report shall begin with the date of issuance of the operating license.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 of this Environmental Protection Plan for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous nonradiological environmental monitoring reports, and an assessment of the observed impacts of the ~~facility~~ plant operation on the environment. If harmful effects or evidence of trends towards irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of action to alleviate the problem.

The Annual Environmental Operating Report shall also include:

(a) A list of EPP noncompliances and the corrective actions taken to remedy them.

(b) A list of all changes in ~~facility~~ station design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental issue.

(c) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of nonroutine event. The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and ~~facility~~ plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided a copy of such report at the time it is submitted to the other agency.

Enclosure Attachment 2 to

HDI PNP 2023-030

Page Change Instructions and Retyped Pages for the Palisades Plant

Renewed Facility License DPR-20,

Appendix A Technical Specifications,

and

Appendix B Environmental Protection Plan

Note, references to "HDI" are replaced by bracketed Palisades Energy, LLC, or Palisades Energy (e.g. [Palisades Energy]) to reflect the change in operating authority per license transfer application conforming amendments.

251 pages follow

Page Change Instructions

ATTACHMENT TO LICENSE AMENDMENT NO. XXX

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Remove the following pages of Palisades Plant Renewed Facility Operating License and replace with the attached revised Palisades Plant Renewed Facility License. The revised pages are identified by amendment number and contains a line in the margin indicating the area of change.

REMOVE	INSERT
Pages 1 through 7	Pages 1 through 8

Remove the following pages of Appendix A, Permanently Defueled Technical Specifications, and replace with the attached revised pages. The revised pages are identified by amendment number and contain a line in the margin indicating the area of change.

REMOVE	INSERT
TS Title Page	TS Title Page
Page i	None
Pages 1.1-1	Page 1.1-1 through 1.1-7
Pages 1.2-1 through 1.2-2	Pages 1.2-1 through 1.2-3
Pages 1.3-1 through 1.3-2	Pages 1.3-1 through 1.3-12
Pages 1.4-1 through 1.4-3	Pages 1.4-1 through 1.4-8
Page 2.0-1	Page 2.0-1
Pages 3.0-1 through 3.0-2	Pages 3.0-1 through 3.0-6
New	Pages 3.1.1-1 through 3.7.13-1
Page 3.7.14-1	Page 3.7.14-1
Page 3.7.15-1	Page 3.7.15-1
Page 3.7.16-1	Page 3.7.16-1
New	Pages 3.7.17-1 through 3.9.6-1
Pages 4.0-1 through 4.0-5	Pages 4.0-1 through 4.0-6
Pages 5.0-7 through 5.0-15	Pages 5.0-7 through 5.0-30

Remove the following pages of Appendix B, Environmental Protection Plan, and replace with the attached revised pages. The revised pages are identified by amendment number and contain a line in the margin indicating the area of change.

REMOVE	INSERT
Cover Sheet	Cover Sheet
Table of Contents	Table of Contents
Page 1-1	Page 1-1
Page 2-1	Page 2-1
Pages 3-1 through 3-3	Pages 3-1 through 3-3
Page 4-1	Page 4-1
Pages 5-1 through 5-4	Pages 5-1 through 5-4

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HOLTEC PALISADES, LLC

[Palisades Energy], LLC

DOCKET NO. 50-255

PALISADES NUCLEAR PLANT

RENEWED FACILITY OPERATING LICENSE

Renewed License No. DPR-20

1. The Nuclear Regulatory Commission (NRC or the Commission) having previously made the findings set forth in Operating License No. DPR-20, dated February 21, 1991, has now found that:

   A. The application for Renewed Operating License No. DPR-20 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. [deleted];

   C. Actions have been identified and have been or will be taken with respect to: (1) managing the effects of aging on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1) during the period of extended operation, and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3 for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations;

D. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

E. There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

F. Holtec Palisades, LLC (Holtec Palisades) is financially qualified and [Palisades Energy, LLC (Palisades Energy)] is financially and technically qualified to engage in the activities authorized by this renewed operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

G. Holtec Palisades and [Palisades Energy] have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations;

H. The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public;

I. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this renewed Facility Operating License No. DPR-20, subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 (formerly Appendix D to Part 50), of the Commission's regulations and all applicable requirements have been satisfied; and

J. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this renewed operating license will be in accordance with 10 CFR Parts 30, 40, and 70.

2. Renewed Facility Operating License No. DPR-20 is hereby issued to Holtec Palisades and [Palisades Energy] as follows:

A. This renewed license applies to the Palisades Plant, a pressurized light water moderated and cooled reactor and electrical generating equipment (the facility). The facility is located in Van Buren County, Michigan, and is described in the Palisades Plant Updated Final Safety Analysis Report, as supplemented and amended, and in the Palisades Plant Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) Holtec Palisades to possess and use, and (b) [Palisades Energy, LLC (Palisades Energy)] to possess, use and operate the facility as a utilization facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;

(2) [Palisades Energy], pursuant to the Act and 10 CFR Parts 40 and 70, to receive , possess and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) [Palisades Energy], pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use byproduct, source and special nuclear material as sealed sources for reactor startup, reactor instrumentation radiation monitoring equipment calibration, and that were used as fission detectors in amounts as required;

(4) [Palisades Energy], pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) [Palisades Energy], pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act; to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) [Palisades Energy] is authorized to operate the facility at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. XXX, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. [Palisades Energy] shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Fire Protection</u>

[Palisades Energy] shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment requests dated December 12, 2012, November 1, 2017, November 1, 2018, and March 8, 2019, as supplemented by letters dated February 21, 2013, September 30, 2013, October 24, 2013, December 2, 2013, April 2, 2014, May 7, 2014, June 17, 2014, August 14, 2014, November 4, 2014, December 18, 2014, January 24, 2018, and May 28, 2019, as approved in the safety evaluations dated February 27, 2015, February 27, 2018, and August 20, 2019. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-8}/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) Other Changes that May Be Made Without Prior NRC Approval

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated February 27, 2015, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as

specified by 2, below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-2, "Plant Modifications Committed," of Entergy Nuclear Operations, Inc. (ENO) letter PNP 2019-028 dated May 28, 2019, to complete the transition to full compliance with 10 CFR 50.48(c) before the end of the refueling outage following the fourth full operating cycle after NRC approval. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Table S-3, "Implementation Items," of ENO letter PNP 2014-097 dated November 4, 2014, within six months after NRC approval, or six months after a refueling outage if in progress at the time of approval with the exception of Implementation Items 3 and 8 which will be completed once the related modifications are installed and validated in the PRA model.

(4) [deleted]

(5) [deleted]

(6) Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

a. Fire fighting response strategy with the following elements:
   1. Pre-defined coordinated fire response strategy and guidance
   2. Assessment of mutual aid fire fighting assets
   3. Designated staging areas for equipment and materials
   4. Command and control
   5. Training of response personnel

b. Operations to mitigate fuel damage considering the following:
   1. Protection and use of personnel assets
   2. Communications
   3. Minimizing fire spread
   4. Procedures for implementing integrated fire response strategy
   5. Identification of readily-available pre-staged equipment
   6. Training on integrated fire response strategy
   7. Spent fuel pool mitigation measures

c. Actions to minimize release to include consideration of:
   1. Water spray scrubbing
   2. Dose to onsite responders

(7) [deleted]

(8) Amendment 257 authorizes the implementation of 10 CFR 50.61a in lieu of 10 CFR 50.61.

D. The facility has been granted certain exemptions from Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors." This section contains leakage test requirements, scheduled and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted in a letter dated December 6, 1989.

These exemptions granted pursuant to 10 CFR 50.12, are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. [Palisades Energy] shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Palisades Nuclear Plant Physical Security Plan."

[Palisades Energy] shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Palisades CSP was approved by License Amendment No. 243 as supplemented by changes approved by License Amendment Nos. 248, 253, 259, and 264.

F. [deleted]

G. Holtec Palisades and [Palisades Energy] shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. [deleted]

I. [deleted]

J. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal scheduled, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

K. This license is effective as of the date of issuance and shall expire at midnight March 24, 2031.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A –Technical Specifications

2. Appendix B – Environmental Protection Plan

Date of Issuance: January 17, 2007

PALISADES PLANT

RENEWED FACILITY OPERATING LICENSE DPR-20

APPENDIX A

TECHNICAL SPECIFICATIONS

As Amended through Amendment No. XXX

1.0 USE AND APPLICATION

1.1 Definitions

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AVERAGE DISINTEGRATION ENERGY - $\bar{E}$	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the primary coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
AXIAL OFFSET (AO)	AO shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core (determined using the incore monitoring system).
AXIAL SHAPE INDEX (ASI)	ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core (determined using the excore monitoring system).
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training and retraining program required by Specification 5.3.2.

1.1 Definitions

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST.

Calibration of instrument channels with Resistance Temperature Detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

Whenever a RTD or thermocouple sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog and bistable channels - the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY;

b. Digital channels - the use of diagnostic programs to test digital hardware and the injection of simulated process data into the channel to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY

1.1 Definitions

CHANNEL FUNCTIONAL TEST (continued)	The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or control rods within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion,” 1989; (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

1.1 Definitions

LEAKAGE	LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE, such as that from pump seals or valve packing (except Primary Coolant Pump seal water leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known not to interfere with the operation of leakage detection systems and not to be pressure boundary LEAKAGE; and

3. Primary Coolant System (PCS) LEAKAGE through a Steam Generator to the Secondary System (primary to secondary LEAKAGE).

b. <u>Unidentified LEAKAGE</u>

All LEAKAGE (except Primary Coolant Pump seal leakoff) that is not identified LEAKAGE;

c. <u>Pressure Boundary LEAKAGE</u>

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in a PCS component body, pipe wall, or vessel wall.

MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average primary coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 5.3.1.

1.1 Definitions

OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 13, Initial Tests and Operation, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT (T_q)	T_q shall be the maximum positive ratio of the power generated in any quadrant minus the average quadrant power, to the average quadrant power.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the primary coolant of 2565.4 MWt.
REFUELING BORON CONCENTRATION	REFUELING BORON CONCENTRATION shall be a Primary Coolant System boron concentration of $\geq$ 1720 ppm and sufficient to assure the reactor is subcritical by $\geq$ 5% $\Delta\rho$ with all control rods withdrawn.

1.1 Definitions

SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length control rods (shutdown and regulating) are fully inserted except for the single rod of highest reactivity worth, which is assumed to be fully withdrawn. However, with all full length control rods verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any full length control rods not capable of being fully inserted, the reactivity worth of these rods must be accounted for in the determination of SDM; and b. There is no change in part length rod position
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the primary coolant.
TOTAL RADIAL PEAKING FACTOR (F_R^T)	F_R^T shall be the maximum ratio of the individual fuel pin power to the core average pin power integrated over the total core height, including tilt.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE PRIMARY COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 300
4	Hot Shutdown[b]	< 0.99	NA	$300 > T_{ave} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

1.2 Logical Connectors

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the plant. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the plant is in a MODE or specified condition stated in the Applicability of the LCO.

Unless otherwise specified, the Completion Time begins when a senior licensed operator on the operating shift crew with responsibility for plant operations makes the determination that an LCO is not met and an ACTIONS Condition is entered. The "otherwise specified" exceptions are varied, such as a Required Action Note or Surveillance Requirement Note that provides an alternative time to perform specific tasks, such as testing, without starting the Completion Time. While utilizing the Note, should a Condition be applicable for any reason not addressed by the Note, the Completion Time begins. Should the time allowance in the Note be exceeded, the Completion Time begins at that point. The exceptions may also be incorporated into the Completion Time. For example, LCO 3.8.1, "AC Sources - Operating," Required Action B.2, requires declaring required feature(s) supported by an inoperable diesel generator, inoperable when the redundant required feature(s) are inoperable. The Completion Time states, "4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)." In this case the Completion Time does not begin until the conditions in the Completion Time are satisfied.

Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the plant is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the

DESCRIPTION (continued)

associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the discovery of the situation that required entry into the Condition, unless otherwise specified.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition, unless otherwise specified.

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of

1.3 Completion Times

DESCRIPTION (continued)

this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B.

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	12 hours 12 hours

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 30 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

---NOTE---
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

1.3 Completion Times

EXAMPLE (continued)

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated. AND A.2 Restore subsystem to OPERABLE status.	1 hour AND Once per 8 hours thereafter 72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered.

1.3 Completion Times

EXAMPLES	EXAMPLE 1.3-7 (continued) The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are “otherwise stated” conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only “required” when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met” and “performed” in these instances conveys specific meanings. A Surveillance is "met” only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being “performed,” constitutes a Surveillance not "met.” “Performance” refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

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1.4 Frequency

DESCRIPTION (continued)

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the plant is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the plant is not in a Mode or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE---------------------------- Not required to be performed until 12 hours after ≥ 25% RTP. --- Perform channel adjustment.	7 days

The interval continues, whether or not the plant operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." The interval continues, whether or not the plant operation is < 25% RTP between performances. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the plant reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE---------------------------- Only required to be met in MODE 1. --- Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the plant is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the plant was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------------NOTE--------------------------- Only required to be performed in MODE 1. --- Perform complete cycle of the valve.	 7 days

The interval continues, whether or not the plant operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the plant reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------------NOTE--------------------------- Not required to be met in MODE 3. --- Verify parameter is within limits.	24 hours

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the plant is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the plant was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained at or above the following DNB correlation safety limits:

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

2.1.1.2 In MODES 1 and 2, the peak Linear Heat Rate (LHR) (adjusted for fuel rod dynamics) shall be maintained at ≤ 21.0 kW/ft.

2.1.2 Primary Coolant System (PCS) Pressure SL

In MODES 1, 2, 3, 4, 5, and 6, the PCS pressure shall be maintained at ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, 5, or 6, restore compliance within 5 minutes.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1

LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9.

LCO 3.0.2

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3

When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the plant shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the plant, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 31 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

LCO 3.0.4 (continued)

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate (exceptions to this Specification are stated in the individual Specifications); or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the plant.

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7 Special Test Exception (STE) LCOs in each applicable LCO section allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

LCO 3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

LCO 3.0.9 (continued)

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

3.0 SR APPLICABILITY

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the plant.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODE 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM to be within limits.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Balance

LCO 3.1.2 The core reactivity balance shall be within ± 1% $\Delta\rho$ of predicted values.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity balance not within limit.	A.1 Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation. AND A.2 Establish appropriate operating restrictions and SRs.	7 days 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	------------------------------NOTE---------------------------- The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 Effective Full Power Days (EFPD) after each fuel loading. --- Verify overall core reactivity balance is within ± 1% $\Delta\rho$ of predicted values.	Prior to entering MODE 1 after each fuel loading <u>AND</u> ----------NOTE--------- Only required after initial 60 EFPD --------------------------- In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained less positive than 0.5 E-4 $\Delta\rho/°F$ at ≤ 2% RATED THERMAL POWER (RTP).

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limits.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MTC is less positive than 0.5 E-4 $\Delta\rho/°F$ at ≤ 2% RTP.	Prior to exceeding 2% RTP after each fuel loading

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Alignment

LCO 3.1.4 All control rods, including their position indication channels, shall be OPERABLE and aligned to within 8 inches of all other rods in their respective group, and the control rod position deviation alarm shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel of rod position indication inoperable for one or more control rods.	A.1 Perform SR 3.1.4.1 (rod position verification).	Once within 15 minutes following any rod motion in that group
B. Rod position deviation alarm inoperable.	B.1 Perform SR 3.1.4.1 (rod position verification).	Once within 15 minutes of movement of any control rod
C. One control rod misaligned by > 8 inches.	C.1 Perform SR 3.2.2.1 (peaking factor verification).	2 hours
	OR	
	C.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
D. One full-length control rod immovable, but trippable.	D.1 Restore control rod to OPERABLE status.	Prior to entering MODE 2 following next MODE 3 entry

ACTIONS

E. Required Action and associated Completion Time not met. <u>OR</u> One or more control rods inoperable for reasons other than Condition D. <u>OR</u> Two or more control rods misaligned by > 8 inches. <u>OR</u> Both rod position indication channels inoperable for one or more control rods.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify the position of each control rod to be within 8 inches of all other control rods in its group.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.2	Perform a CHANNEL CHECK of the control rod position indication channels.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.3	Verify control rod freedom of movement by moving each individual full-length control rod that is not fully inserted into the reactor core ≥ 6 inches in either direction.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.4	Verify the rod position deviation alarm is OPERABLE.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.5	Perform a CHANNEL CALIBRATION of the control rod position indication channels.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.6	Verify each full-length control rod drop time is ≤ 2.5 seconds.	Prior to reactor criticality, after each reinstallation of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown and Part-Length Control Rod Group Insertion Limits

LCO 3.1.5 All shutdown and part-length rod groups shall be withdrawn to $\geq$ 128 inches.

APPLICABILITY: MODE 1,
MODE 2 with any regulating rod withdrawn above 5 inches.

---NOTE---
This LCO is not applicable while performing SR 3.1.4.3 (rod exercise test).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown or part-length rods not within limit.	A.1 Declare affected control rod(s) inoperable and enter the applicable Conditions and Required Actions of LCO 3.1.4.	Immediately
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown and part-length rod group is withdrawn $\geq$ 128 inches.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Regulating Rod Group Position Limits

LCO 3.1.6 The Power Dependent Insertion Limit (PDIL) alarm circuit and the Control Rod Out Of Sequence (CROOS) alarm circuit shall be OPERABLE, and the regulating rod groups shall be limited to the withdrawal sequence, overlap, and insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

--NOTE---
This LCO is not applicable while performing SR 3.1.4.3 (rod exercise test).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Regulating rod groups inserted beyond the insertion limit.	A.1 Restore regulating rod groups to within limits.	2 hours
	<u>OR</u>	
	A.2 Reduce THERMAL POWER to less than or equal to the fraction of RTP allowed by the regulating rod group position and insertion limits specified in the COLR.	2 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Regulating rod groups not within sequence or overlap limits.	B.1	Restore regulating rod groups to within appropriate sequence and overlap limits.	2 hours
C. PDIL or CROOS alarm circuit inoperable.	C.1	Perform SR 3.1.6.1 (group position verification).	Once within 15 minutes following any rod motion
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify each regulating rod group is within its withdrawal sequence, overlap, and insertion limits.	In accordance with the Surveillance Frequency Control Program
SR 3.1.6.2	Verify PDIL alarm circuit is OPERABLE.	In accordance with the Surveillance Frequency Control Program
SR 3.1.6.3	Verify CROOS alarm circuit is OPERABLE.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Special Test Exceptions (STE)

LCO 3.1.7 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.4, "Control Rod Alignment";
LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits";
LCO 3.1.6, "Regulating Rod Group Position Limits"; and
LCO 3.4.2, "PCS Minimum Temperature for Criticality"

may be suspended, 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: MODE 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Reduce THERMAL POWER to within limit.	15 minutes
B. Shutdown reactivity not within limit.	B.1 Initiate boration to restore shutdown reactivity to within limit.	15 minutes
C. T_{ave} not within limit.	C.1 Restore T_{ave} to within limit.	15 minutes

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Suspend PHYSICS TESTS.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify THERMAL POWER is ≤ 2% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.2	Verify T_{ave} is ≥ 500°F.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.3	Verify ≥ 1% shutdown reactivity is available for trip insertion.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Linear Heat Rate (LHR)

LCO 3.2.1 LHR shall be within the limits specified in the COLR, and the Incore Alarm System or Excore Monitoring System shall be OPERABLE to monitor LHR.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LHR, as determined by the automatic Incore Alarm System, not within limits specified in the COLR, as indicated by four or more coincident incore channels. <u>OR</u> LHR, as determined by the Excore Monitoring System, not within limits specified in the COLR. <u>OR</u> LHR, as determined by manual incore detector readings, not within limits specified in the COLR.	A.1 Restore LHR to within limits.	1 hour

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Incore Alarm and Excore Monitoring Systems inoperable for monitoring LHR.	B.1 Reduce THERMAL POWER to ≤ 85% RTP. <u>AND</u> B.2 Verify LHR is within limits using manual incore readings.	2 hours 4 hours <u>AND</u> Once per 2 hours thereafter
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to ≤ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	------------------------------NOTE-------------------------- Only required to be met when the Incore Alarm System is being used to monitor LHR. -- Verify LHR is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	------------------------------NOTE---------------------------- Only required to be met when the Incore Alarm System is being used to monitor LHR. --- Adjust incore alarm setpoints based on a measured power distribution.	Prior to operation > 50% RTP after each fuel loading <u>AND</u> In accordance with the Surveillance Frequency Control Program
SR 3.2.1.3	------------------------------NOTE---------------------------- Only required to be met when the Excore Monitoring System is being used to monitor LHR. --- Verify measured ASI has been within 0.05 of target ASI for last 24 hours.	Prior to each initial use of Excore Monitoring System to monitor LHR
SR 3.2.1.4	------------------------------NOTE---------------------------- Only required to be met when the Excore Monitoring System is being used to monitor LHR. --- Verify THERMAL POWER is less than the APL.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.5	-------------------------------NOTE--------------------------- Only required to be met when the Excore Monitoring System is being used to monitor LHR. --- Verify measured ASI is within 0.05 of target ASI.	In accordance with the Surveillance Frequency Control Program
SR 3.2.1.6	-------------------------------NOTE--------------------------- Only required to be met when the Excore Monitoring System is being used to monitor LHR. --- Verify $T_q \leq 0.03$.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.2 TOTAL RADIAL PEAKING FACTOR (F_R^T)

LCO 3.2.2 F_R^T shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. F_R^T not within limits specified in the COLR.	A.1 Restore F_R^T to within limits.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify F_R^T is within limits specified in the COLR.	Prior to operation > 50% RTP after each fuel loading <u>AND</u> In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.3 QUADRANT POWER TILT (T_q)

LCO 3.2.3 T_q shall be $\leq$ 0.05.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_q > 0.05.	A.1 Verify F_R^T is within the limits of LCO 3.2.2, "TOTAL RADIAL PEAKING FACTOR ".	2 hours <u>AND</u> Once per 8 hours thereafter
B. T_q > 0.10.	B.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
C. Required Action and associated Completion Time not met. <u>OR</u> T_q > 0.15.	C.1 Reduce THERMAL POWER to $\leq$ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify T_q is $\leq$ 0.05.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.4 AXIAL SHAPE INDEX (ASI)

LCO 3.2.4 The ASI shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ASI not within limits specified in COLR.	A.1 Restore ASI to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	Verify ASI is within limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.1 Reactor Protective System (RPS) Instrumentation

LCO 3.3.1 Four RPS trip units, associated instrument channels, and associated Zero Power Mode (ZPM) Bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE-------------- Not applicable to High Startup Rate, Loss of Load, or ZPM Bypass Removal Functions. ---------------------------------- One or more Functions with one RPS trip unit or associated instrument channel inoperable.	A.1 Place affected trip unit in trip.	7 days
B. One High Startup Rate trip unit or associated instrument channel inoperable.	B.1 Restore trip unit and associated instrument channel to OPERABLE status.	Prior to entering MODE 2 from MODE 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One Loss of Load trip unit or associated instrument channel inoperable.	C.1 Restore trip unit and associated instrument channel to OPERABLE status.	Prior to increasing THERMAL POWER to ≥ 17% RTP following entry into MODE 3
D. One or more ZPM Bypass Removal channels inoperable.	D.1 Remove the affected ZPM Bypasses.	Immediately
	<u>OR</u>	
	D.2 Declare affected trip units inoperable.	Immediately
E. -------------NOTE-------------- Not applicable to ZPM Bypass Removal Function. ----------------------------------- One or more Functions with two RPS trip units or associated instrument channels inoperable.	E.1 Place one trip unit in trip.	1 hour
	<u>AND</u>	
	-----------------NOTE------------------- Not applicable to High Startup Rate or Loss of Load Functions. -- E.2 Restore one trip unit and associated instrument channel to OPERABLE status.	7 days
F. Two power range channels inoperable.	F.1 Restrict THERMAL POWER to ≤ 70% RTP.	2 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. Required Action and associated Completion Time not met. OR Control room ambient air temperature > 90ºF.	G.1	Be in MODE 3.	6 hours
	AND		
	G.2.1	Verify no more than one full-length control rod is capable of being withdrawn.	6 hours
	OR		
	G.2.2	Verify PCS boron concentration is at REFUELING BORON CONCENTRATION.	6 hours

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.1-1 to determine which SR shall be performed for each Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform a CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.2	Verify control room temperature is ≤ 90°F.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.1.3	-----------------------------NOTE----------------------------- Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. --- Perform calibration (heat balance only) and adjust the power range excore and ΔT power channels to agree with calorimetric calculation if the absolute difference is ≥ 1.5%.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.4	-----------------------------NOTE----------------------------- Not required to be performed until 12 hours after THERMAL POWER is ≥ 25% RTP. --- Calibrate the power range excore channels using the incore detectors.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.5	Perform a CHANNEL FUNCTIONAL TEST and verify the Thermal Margin Monitor Constants.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.6	Perform a calibration check of the power range excore channels with a test signal.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.7	Perform a CHANNEL FUNCTIONAL TEST of High Startup Rate and Loss of Load Functions.	Once within 7 days prior to each reactor startup

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	-----------------------------NOTE----------------------------- Neutron detectors are excluded from the CHANNEL CALIBRATION. --- Perform a CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

Table 3.3.1-1 (page 1 of 2)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable High Power Trip	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.8	≤ 15% RTP above current THERMAL POWER with a minimum of ≤ 30% RTP and a maximum of ≤ 109.4% RTP
2. High Startup Rate Trip[(b)]	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.8	NA
3. Low Primary Coolant System Flow Trip[(c)]	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 95%
4. Low Steam Generator A Level Trip	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 25.9% narrow range
5. Low Steam Generator B Level Trip	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 25.9% narrow range
6. Low Steam Generator A Pressure Trip[(c)]	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 500 psia
7. Low Steam Generator B Pressure Trip[(c)]	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 500 psia
8. High Pressurizer Pressure Trip	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≤ 2255 psia

(a) With more than one full-length control rod capable of being withdrawn and PCS boron concentration less than REFUELING BORON CONCENTRATION.

(b) Trip may be bypassed when Wide Range Power is < 1E-4% RTP or when THERMAL POWER is > 13% RTP.

(c) Trips may be bypassed when Wide Range Power is < 1E-4% RTP. Bypass shall be automatically removed when Wide Range Power is ≥ 1E-4% RTP.

Table 3.3.1-1 (page 2 of 2)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Thermal Margin/ Low Pressure Trip[c]	1,2,3[a],4[a],5[a]	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.8	Table 3.3.1-2
10. Loss of Load Trip	1[d]	SR 3.3.1.7 SR 3.3.1.8	NA
11. Containment High Pressure Trip	1,2,3[a],4[a],5[a]	SR 3.3.1.5 SR 3.3.1.8	≤ 3.70 psig
12. Zero Power Mode Bypass Automatic Removal	1,2,3[a],4[a],5[a]	SR 3.3.1.8	NA

(a) With more than one full-length control rod capable of being withdrawn and PCS boron concentration less than REFUELING BORON CONCENTRATION.

(c) Trips may be bypassed when Wide Range Power is < 1E-4% RTP. Bypass shall be automatically removed when Wide Range Power is ≥ 1E-4% RTP.

(d) When THERMAL POWER is ≥ 17% RTP.

Table 3.3.1-2 (page 1 of 1)
Thermal Margin/Low Pressure Trip Function Allowable Value

The Allowable Value for the Thermal Margin/Low Pressure Trip, P_{trip}, is the higher of two values, P_{min} and P_{var}, both in psia:

$P_{min} = 1750$
$P_{var} = 2012(QA)(QR_1) + 17.0(T_{in}) - 9559$

Where:

$QA = - 0.720(ASI) + 1.028$;	when $- 0.628 \leq ASI < - 0.100$
$QA = - 0.333(ASI) + 1.067$;	when $- 0.100 \leq ASI < + 0.200$
$QA = + 0.375(ASI) + 0.925$;	when $+ 0.200 \leq ASI \leq + 0.565$
ASI = Measured ASI	when $Q \geq 0.0625$
ASI = 0.0	when $Q < 0.0625$
$QR_1 = 0.412(Q) + 0.588$;	when $Q \leq 1.0$
$QR_1 = Q$;	when $Q > 1.0$

Q = THERMAL POWER/RATED THERMAL POWER

T_{in} = Maximum primary coolant inlet temperature, in °F

ASI, T_{in}, and Q are the existing values as measured by the associated instrument channel.

3.3 INSTRUMENTATION

3.3.2 Reactor Protective System (RPS) Logic and Trip Initiation

LCO 3.3.2 Six channels of RPS Matrix Logic, four channels of RPS Trip Initiation Logic, and two channels of RPS Manual Trip shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5, with more than one full-length control rod capable of being withdrawn and Primary Coolant System (PCS) boron concentration less than REFUELING BORON CONCENTRATION.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Matrix Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours
B. One channel of Trip Initiation Logic inoperable.	B.1 De-energize the affected clutch power supplies.	1 hour
C. One channel of Manual Trip inoperable.	C.1 Restore channel to OPERABLE status.	Prior to entering MODE 2 from MODE 3
D. Two channels of Trip Initiation Logic affecting the same trip leg inoperable.	D.1 De-energize the affected clutch power supplies.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Required Action and associated Completion Time not met. <u>OR</u> One or more Functions with two or more Manual Trip, Matrix Logic or Trip Initiation Logic channels inoperable for reasons other than Condition D.	E.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.2.1	Verify no more than one full-length control rod is capable of being withdrawn.	6 hours
	<u>OR</u>		
	E.2.2	Verify PCS boron concentration is at REFUELING BORON CONCENTRATION.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform a CHANNEL FUNCTIONAL TEST on each RPS Matrix Logic channel and each RPS Trip Initiation Logic channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.2	Perform a CHANNEL FUNCTIONAL TEST on each RPS Manual Trip channel.	Once within 7 days prior to each reactor startup

3.3 INSTRUMENTATION

3.3.3 Engineered Safety Features (ESF) Instrumentation

LCO 3.3.3 Four ESF bistables and associated instrument channels for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: As specified in Table 3.3.3-1.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------------NOTE--------------- Not applicable to RAS. ------------------------------------- One or more Functions with one ESF bistable or associated instrument channel inoperable.	A.1 Place affected bistable in trip.	7 days
B. ---------------NOTE--------------- Not applicable to RAS. ------------------------------------- One or more Functions with two ESF bistables or associated instrument channels inoperable.	B.1 Place one bistable in trip. <u>AND</u> B.2 Restore one bistable and associated instrument channel to OPERABLE status.	8 hours 7 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One RAS bistable or associated instrument channel inoperable.	C.1 Bypass affected bistable. AND C.2 Restore bistable and associated instrument channel to OPERABLE status.	8 hours 7 days
D. Required Action and associated Completion Time not met for Functions 1, 2, 3, 4, or 7.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 30 hours
E. Required Action and associated Completion Time not met for Functions 5 or 6.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

--NOTE--

Refer to Table 3.3.3-1 to determine which SR shall be performed for each Function.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform a CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.3.2	Perform a CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.3.3	Perform a CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

Table 3.3.3-1 (page 1 of 2)
Engineered Safety Features Instrumentation

FUNCTION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection Signal (SIS)			
a. Pressurizer Low Pressure	1,2,3	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 1593 psia
2. Steam Generator Low Pressure Signal (SGLP)			
a. Steam Generator A Low Pressure	1,2[(a)],3[(a)]	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 500 psia
b. Steam Generator B Low Pressure	1,2[(a)],3[(a)]	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 500 psia
3. Recirculation Actuation Signal (RAS)			
a. SIRWT Low Level	1,2,3	SR 3.3.3.3	≥ 21 inches and ≤ 27 inches above tank bottom
4. Auxiliary Feedwater Actuation Signal (AFAS)			
a. Steam Generator A Low Level	1,2,3	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 25.9% narrow range
b. Steam Generator B Low Level	1,2,3	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 25.9% narrow range

(a) Not required to be OPERABLE when 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.

Table 3.3.3-1 (page 2 of 2)
Engineered Safety Features Instrumentation

FUNCTION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Containment High Pressure (CHP)			
a. Containment High Pressure — Left Train	1,2,3,4	SR 3.3.3.2 SR 3.3.3.3	≥ 3.7 psig and ≤ 4.3 psig
b. Containment High Pressure — Right Train	1,2,3,4	SR 3.3.3.2 SR 3.3.3.3	≥ 3.7 psig and ≤ 4.3 psig
6. Containment High Radiation Signal (CHR)			
a. Containment High Radiation	1,2,3,4	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≤ 20 R/hour
7. Automatic Bypass Removals			
a. Pressurizer Low Pressure Bypass	1,2,3	SR 3.3.3.3	≤ 1700 psia
b. Steam Generator A Low Pressure Bypass	1,2[(a)],3[(a)]	SR 3.3.3.3	≤ 565 psia
c. Steam Generator B Low Pressure Bypass	1,2[(a)],3[(a)]	SR 3.3.3.3	≤ 565 psia

(a) Not required to be OPERABLE when 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.

3.3 INSTRUMENTATION

3.3.4 Engineered Safety Features (ESF) Logic and Manual Initiation

LCO 3.3.4 Two ESF Manual Initiation and two ESF Actuation Logic channels and associated bypass removal channels shall be OPERABLE for each ESF Function specified in Table 3.3.4-1.

APPLICABILITY: According to Table 3.3.4-1.

ACTIONS

--NOTE--

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one Manual Initiation, Bypass Removal, or Actuation Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours
B. One or more Functions with two Manual Initiation, Bypass Removal, or Actuation Logic channels inoperable for Functions 1, 2, 3, or 4. <u>OR</u> Required Action and associated Completion Time of Condition A not met for Functions 1, 2, 3, or 4.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 30 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with two Manual Initiation, or Actuation Logic channels inoperable for Functions 5 or 6. OR Required Action and associated Completion Time of Condition A not met for Functions 5 or 6.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform functional test of each SIS actuation channel normal and standby power functions.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.2	Perform a CHANNEL FUNCTIONAL TEST of each AFAS actuation logic channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.3	Perform a CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

Table 3.3.4-1 (page 1 of 1)
Engineered Safety Features Actuation Logic and Manual Initiation

	FUNCTION	APPLICABLE MODES
1.	Safety Injection Signal (SIS)[(a)]	1,2,3
2.	Steam Generator Low Pressure Signal (SGLP)[(b)(c)]	1,2[(d)],3[(d)]
3.	Recirculation Actuation Signal (RAS)	1,2,3
4.	Auxiliary Feedwater Actuation Signal (AFAS)	1,2,3
5.	Containment High Pressure Signal (CHP)[(c)]	1,2,3,4
6.	Containment High Radiation Signal (CHR)	1,2,3,4

(a) SIS actuation by Pressurizer Low Pressure may be manually bypassed when pressurizer pressure is ≤ 1700 psia. The bypass shall be automatically removed whenever pressurizer pressure is > 1700 psia.

(b) SGLP actuation may be manually bypassed when SG pressure is ≤ 565 psia. The bypass shall be automatically removed whenever steam generator pressure is > 565 psia.

(c) Manual Initiation may be achieved by individual component controls.

(d) Not required to be OPERABLE when 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.

3.3 INSTRUMENTATION

3.3.5 Diesel Generator (DG) - Undervoltage Start (UV Start)

LCO 3.3.5 Three channels of Loss of Voltage Function and three channels of Degraded Voltage Function auto-initiation instrumentation and associated logic channels for each DG shall be OPERABLE.

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel per DG inoperable.	A.1 Enter applicable Conditions and Required Actions for the associated DG made inoperable by DG - UV Start instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform a CHANNEL FUNCTIONAL TEST on each DG-UV start logic channel.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.2	Perform CHANNEL CALIBRATION on each Loss of Voltage and Degraded Voltage channel with setpoints as follows: a. Degraded Voltage Function ≥ 2187 V and ≤ 2264 V 1. Time delay (degraded voltage sensing relay): ≥ 0.5 seconds and ≤ 0.8 seconds; and 2. Time delay (degraded voltage sensing relay plus time delay relay): ≥ 6.2 seconds and ≤ 7.1 seconds. b. Loss of Voltage Function ≥ 1780 V and ≤ 1940 V Time delay: ≥ 5.45 seconds and ≤ 8.15 seconds at 1400 V.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.6 Refueling Containment High Radiation (CHR) Instrumentation

LCO 3.3.6 Two Refueling CHR Automatic Actuation Function channels and two CHR Manual Actuation Function channels shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable.	A.1 Place the affected channel in trip.	4 hours
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	4 hours
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies within containment.	4 hours
B. One or more Functions with two channels inoperable.	B.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	B.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform a CHANNEL CHECK of each refueling CHR monitor channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.2	Perform a CHANNEL FUNCTIONAL TEST of each refueling CHR monitor channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.3	Perform a CHANNEL FUNCTIONAL TEST of each CHR Manual Initiation channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.4	Perform a CHANNEL CALIBRATION of each refueling CHR monitor channel.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.7 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.7 The PAM instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.6.	Immediately
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (Not Used)			
E. Required Action and associated Completion Time of Condition C not met.	E.1	Enter the Condition referenced in Table 3.3.7-1 for the channel.	Immediately
F. As required by Required Action E.1 and referenced in Table 3.3.7-1.	F.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	F.2	Be in MODE 4.	30 hours
G. As required by Required Action E.1 and referenced in Table 3.3.7-1.	G.1	Initiate action in accordance with Specification 5.6.6.	Immediately

SURVEILLANCE REQUIREMENTS

--NOTE--

These SRs apply to each PAM instrumentation Function in Table 3.3.7-1.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.2	----------------------------NOTE---------------------------- Neutron detectors are excluded from the CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

Table 3.3.7-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION E.1
1.	Primary Coolant System Hot Leg Temperature (wide range)	2	F
2.	Primary Coolant System Cold Leg Temperature (wide range)	2	F
3.	Wide Range Neutron Flux	2	F
4.	Containment Floor Water Level (wide range)	2	F
5.	Subcooled Margin Monitor	2	F
6.	Pressurizer Level (wide range)	2	F
7.	(Deleted)		
8.	Condensate Storage Tank Level	2	F
9.	Primary Coolant System Pressure (wide range)	2	F
10.	Containment Pressure (wide range)	2	F
11.	Steam Generator A Water Level (wide range)	2	F
12.	Steam Generator B Water Level (wide range)	2	F
13.	Steam Generator A Pressure	2	F
14.	Steam Generator B Pressure	2	F
15.	Containment Isolation Valve Position	1 per valve[a]	F
16.	Core Exit Temperature - Quadrant 1	4	F
17.	Core Exit Temperature - Quadrant 2	4	F
18.	Core Exit Temperature - Quadrant 3	4	F
19.	Core Exit Temperature - Quadrant 4	4	F
20.	Reactor Vessel Water Level	2	G
21.	Containment Area Radiation (high range)	2	G

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

3.3 INSTRUMENTATION

3.3.8 Alternate Shutdown System

LCO 3.3.8 The Alternate Shutdown System Functions in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable.	A.1 Restore required Functions to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4.	30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform CHANNEL FUNCTIONAL TEST of the Source Range Neutron Flux Function.	Once within 7 days prior to each reactor startup
SR 3.3.8.2	Verify each required control circuit and transfer switch is capable of performing the intended function.	In accordance with the Surveillance Frequency Control Program
SR 3.3.8.3	-----------------------------NOTES--------------------------- 1. Not required for Functions 16, 17, and 18. 2. Neutron detectors are excluded from the CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION for each required instrumentation channel.	In accordance with the Surveillance Frequency Control Program

Table 3.3.8-1 (page 1 of 1)
Alternate Shutdown System Instrumentation and Controls

	FUNCTION, INSTRUMENT OR CONTROL PARAMATER	REQUIRED CHANNELS
1.	Source Range Neutron Flux	1
2.	Pressurizer Pressure	1
3.	Pressurizer Level	1
4.	Primary Coolant System (PCS) #1 Hot Leg Temperature	1
5.	PCS #2 Hot Leg Temperature	1
6.	PCS #1 Cold Leg Temperature	1
7.	PCS #2 Cold Leg Temperature	1
8.	Steam Generator (SG) A Pressure	1
9.	SG B Pressure	1
10.	SG A Wide Range Level	1
11.	SG B Wide Range Level	1
12.	Safety Injection Refueling Water (SIRW) Tank Level	1
13.	Auxiliary Feedwater (AFW) Flow Indication to SG A	1
14.	AFW Flow Indication to SG B	1
15.	AFW Low Suction Pressure Alarm (P-8B)	1
16.	AFW Pump P-8B Steam Supply Valve Control	1
17.	AFW Flow Control to SG A	1
18.	AFW Flow Control to SG B	1

3.3 INSTRUMENTATION

3.3.9 Neutron Flux Monitoring Channels

LCO 3.3.9 Two channels of neutron flux monitoring instrumentation shall be OPERABLE.

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channel(s) inoperable.	A.1 Suspend all operations involving positive reactivity additions. AND	Immediately
	A.2 Perform SDM verification in accordance with SR 3.1.1.1.	4 hours AND Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.9.2	----------------------------NOTE----------------------------- Neutron detectors are excluded from the CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.10 Engineered Safeguards Room Ventilation (ESRV) Instrumentation

LCO 3.3.10 Two channels of ESRV Instrumentation shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Initiate action to isolate the associated ESRV System.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform a CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.10.2	Perform a CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.10.3	Perform a CHANNEL CALIBRATION. Verify high radiation setpoint on each ESRV Instrumentation radiation monitoring channel is ≤ 2.2E+5 cpm.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

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

LCO 3.4.1 PCS DNB parameters for pressurizer pressure, cold leg temperature, and PCS total flow rate shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer pressure, PCS cold leg temperature, or PCS total flow rate not within limits.	A.1 Restore parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.2	Verify PCS cold leg temperature within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.3	------------------------------NOTE---------------------------- Not required to be performed until 31 EFPD after THERMAL POWER is ≥ 90% RTP. --- Verify PCS total flow rate within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program <u>AND</u> After each plugging of 10 or more steam generator tubes

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.2 PCS Minimum Temperature for Criticality

LCO 3.4.2 Each PCS loop average temperature (T_{ave}) shall be ≥ 525°F.

APPLICABILITY: MODE 1
MODE 2 with $K_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{ave} in one or more PCS loops not within limit.	A.1 Be in MODE 2 with $K_{eff} < 1.0$.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify PCS T_{ave} in each loop $\geq$ 525ºF.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.3 PCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 PCS pressure, PCS temperature, and PCS heatup and cooldown rates shall be maintained within the limits of Figure 3.4.3-1 and Figure 3.4.3-2.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE-------------- Required Action A.2 shall be completed whenever this Condition is entered. ---------------------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. <u>AND</u> A.2 Determine PCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5 with PCS pressure < 270 psia.	6 hours 36 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. --------------NOTE------------- Required Action C.2 shall be completed whenever this Condition is entered. ----------------------------------	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	<u>AND</u>		
Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.2	Determine PCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-------------------------------NOTE--------------------------- Only required to be performed during PCS heatup and cooldown operations. --	
	Verify PCS pressure, PCS temperature, and PCS heatup and cooldown rates are within the limits of Figure 3.4.3-1 and Figure 3.4.3-2.	In accordance with the Surveillance Frequency Control Program

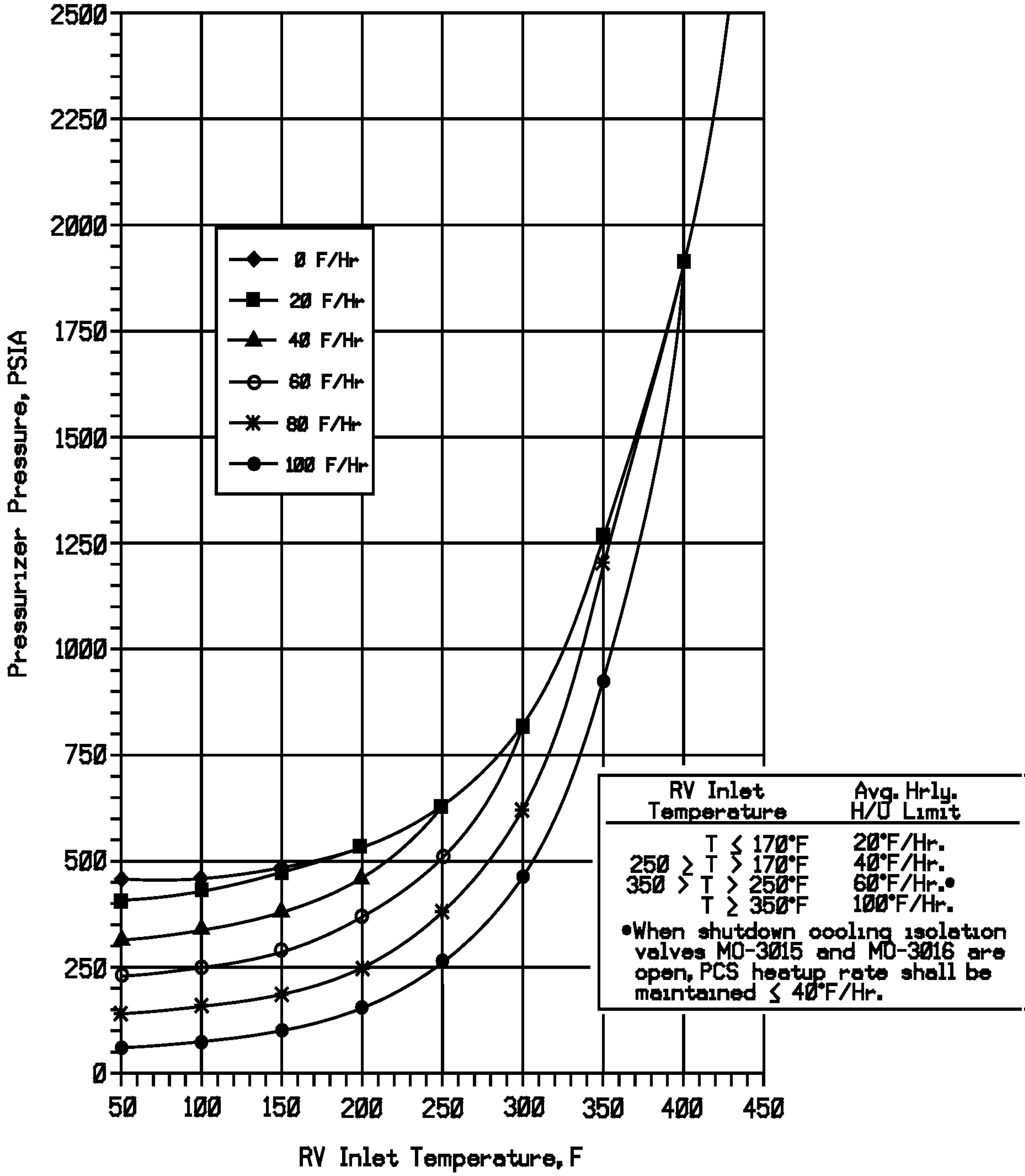
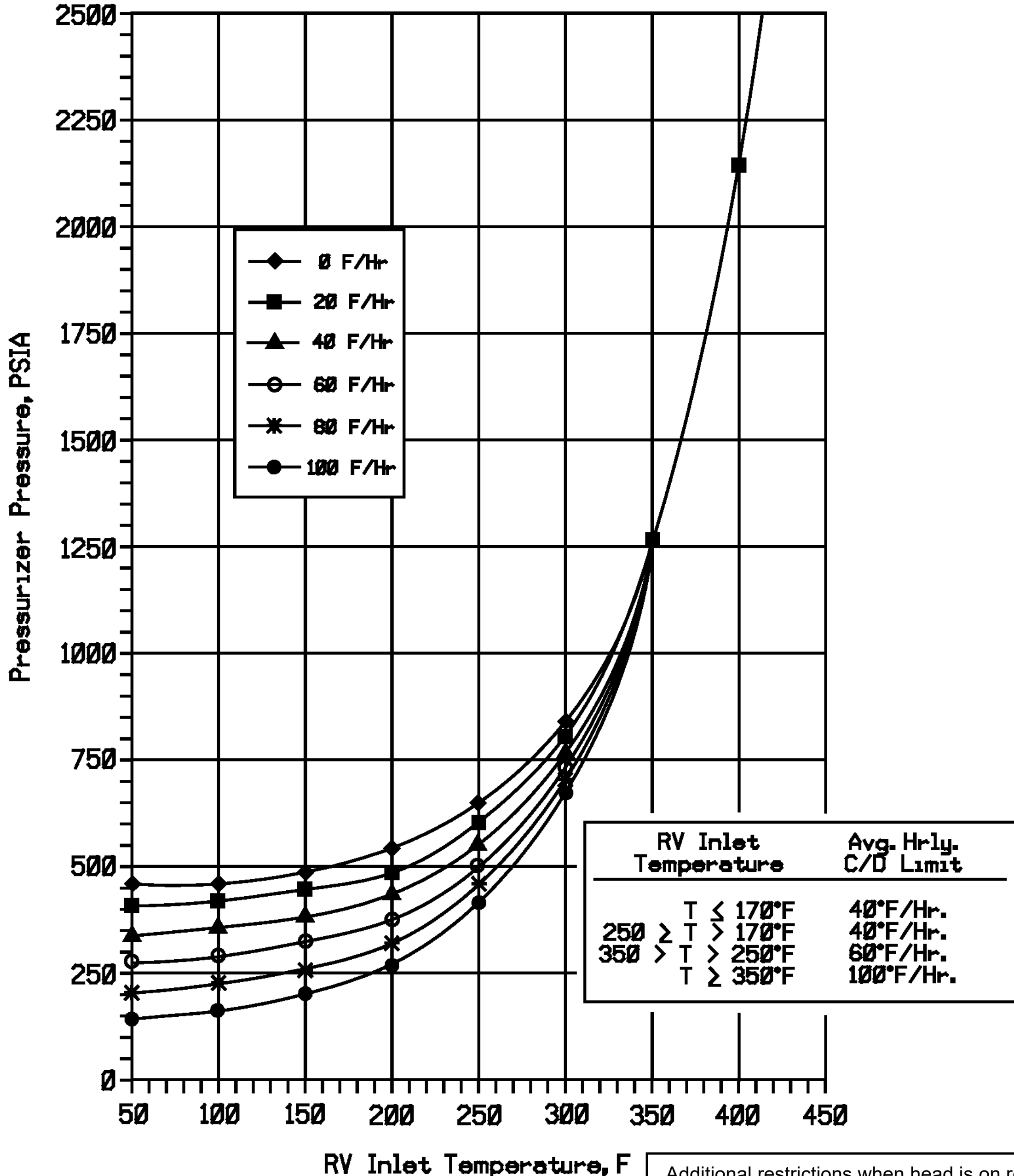


Figure 3.4.3-1 (Page 1 of 1)
Pressure – Temperature Limits for Heatups
Applicable up to 42.1 EFPY



Additional restrictions when head is on reactor vessel:

1. Maintain average core exit temperature:
   135°F ≥ T ≥ 110°F for ≥ 3 hours.

2. Following completion of item 1, maintain average hourly cooldown (C/D) limit of 20°F/hour based on core exit temperature indication.

Figure 3.4.3-2 (Page 1 of 1)
Pressure – Temperature Limits for Cooldown
Applicable up to 42.1 EFPY

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.4 PCS Loops - MODES 1 and 2

LCO 3.4.4 Two PCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each PCS loop is in operation.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.5 PCS Loops - MODE 3

LCO 3.4.5 Two PCS loops shall be OPERABLE and one PCS loop shall be in operation.

---NOTES--

1. All primary coolant pumps may not be in operation for ≤ 1 hour per 8 hour period, provided:

    a. No operations are permitted that would cause reduction of the PCS boron concentration; and

    b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. Forced circulation (starting the first primary coolant pump) shall not be initiated unless one of the following conditions is met:

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

    b. Steam Generator (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

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

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required PCS loop inoperable.	A.1 Restore required PCS loop to OPERABLE status.	72 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	24 hours
C. No PCS loop OPERABLE. <u>OR</u> No PCS loop in operation.	C.1 Suspend all operations involving a reduction of PCS boron concentration. <u>AND</u> C.2 Initiate action to restore one PCS loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required PCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.5.2	Verify secondary side water level in each steam generator ≥ -84%.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.3	Verify correct breaker alignment and indicated power available to the required primary coolant pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.6 PCS Loops - MODE 4

LCO 3.4.6 Two loops or trains consisting of any combination of PCS loops and Shutdown Cooling (SDC) trains shall be OPERABLE, and either:

a. One PCS loop shall be in operation; or

b. One SDC train shall be in operation with ≥ 2810 gpm flow through the reactor core.

---NOTES---

1. All Primary Coolant Pumps (PCPs) and SDC pumps may not be in operation for ≤ 1 hour per 8 hour period, provided:

    a. No operations are permitted that would cause reduction of the PCS boron concentration; and

    b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. Forced circulation (starting the first PCP) shall not be initiated unless one of the following conditions is met:

    a. Steam Generator (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,

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

3. Primary coolant pumps P-50A and P-50B shall not be operated simultaneously.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One PCS loop inoperable. AND Two SDC trains inoperable.	A.1	Initiate action to restore a second PCS loop or one SDC train to OPERABLE status.	Immediately
B. One SDC train inoperable. AND Two PCS loops inoperable.	B.1	Be in MODE 5.	24 hours
C. No PCS loops or SDC trains OPERABLE. OR No PCS loop in operation with SDC flow through the reactor core not within limits.	C.1	Suspend all operations involving reduction of PCS boron concentration.	Immediately
	AND		
	C.2.1	Initiate action to restore one PCS loop to OPERABLE status and operation. OR	Immediately
	C.2.2	Initiate action to restore one SDC train to OPERABLE status and operation with ≥ 2810 gpm flow through the reactor core.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one SDC train is in operation with ≥ 2810 gpm flow through the reactor core, or one PCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.2	Verify secondary side water level in required SG(s) is ≥ -84%.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.3	Verify correct breaker alignment and indicated power available to the required pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.7 PCS Loops - MODE 5, Loops Filled

LCO 3.4.7

One Shutdown Cooling (SDC) train shall be OPERABLE and in operation with $\geq$ 2810 gpm flow through the reactor core, and either:

a. One additional SDC train shall be OPERABLE; or

b. The secondary side water level of each Steam Generator (SG) shall be $\geq$ -84%.

--NOTES--

1. The SDC pump of the train in operation may not be in operation for $\leq$ 1 hour per 8 hour period provided:
   a. No operations are permitted that would cause reduction of the PCS boron concentration; and
   b. Core outlet temperature is maintained at least 10°F below saturation temperature.
2. Both SDC trains may be inoperable for up to 2 hours for surveillance testing or maintenance provided:
   a. One SDC train is providing the required flow through the reactor core;
   b. Core outlet temperature is maintained at least 10°F below saturation temperature; and
   c. Each SG secondary side water level is $\geq$ -84%.
3. Forced circulation (starting the first primary coolant pump) shall not be initiated unless one of the following conditions is met:
   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 $\leq$ 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 $\leq$ 57%.
4. Primary coolant pumps P-50A and P-50B shall not be operated simultaneously.
5. All SDC trains may not be in operation during planned heatup to MODE 4 when at least one PCS loop is in operation.

APPLICABILITY: MODE 5 with PCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SDC train inoperable. <u>AND</u> Any SG with secondary side water level not within limit.	A.1 Initiate action to restore a second SDC train to OPERABLE status. <u>OR</u> A.2 Initiate action to restore SG secondary side water levels to within limits.	Immediately Immediately
B. Two SDC trains inoperable. <u>OR</u> SDC flow through the reactor core not within limits.	B.1 Suspend all operations involving reduction in PCS boron concentration. <u>AND</u> B.2 Initiate action to restore one SDC train to OPERABLE status and operation with $\geq$ 2810 gpm flow through the reactor core.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one SDC train is in operation with $\geq$ 2810 gpm flow through the reactor core.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.2	Verify required SG secondary side water level is $\geq$ - 84%.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.3	Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.8 PCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two Shutdown Cooling (SDC) trains shall be OPERABLE, and either:

a. One SDC train in operation with ≥ 2810 gpm flow through the reactor core; or

b. One SDC train in operation with ≥ 650 gpm flow through the reactor core 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.

---NOTES---

1. All SDC pumps may not be in operation for ≤ 1 hour provided:

   a. No operations are permitted that would cause a reduction of the PCS boron concentration;

   b. Core outlet temperature is maintained > 10°F below saturation temperature; and

   c. No draining operations to further reduce the PCS water volume are permitted.

2. One SDC train may be inoperable for ≤ 2 hours for surveillance testing provided the other SDC train is OPERABLE and in operation.

APPLICABILITY: MODE 5 with PCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SDC train inoperable.	A.1 Initiate action to restore SDC train to OPERABLE status.	Immediately
B. Two SDC trains inoperable. <u>OR</u> SDC flow through the reactor core not within limits.	B.1 Suspend all operations involving reduction of PCS boron concentration. <u>AND</u> B.2 Initiate action to restore one SDC train to OPERABLE status and operation with SDC flow through the reactor core within limit.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	-------------------------------NOTE--------------------------- Only required to be met when complying with LCO 3.4.8.a. -- Verify one SDC train is in operation with $\geq$ 2810 gpm flow through the reactor core.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.4.8.2	-------------------------------NOTE-------------------------- Only required to be met when complying with LCO 3.4.8.b. --- Verify one SDC train is in operation with ≥ 650 gpm flow through the reactor core.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.3	-------------------------------NOTE-------------------------- Only required to be met when complying with LCO 3.4.8.b. --- Verify two of three charging pumps are incapable of reducing the boron concentration in the PCS below the minimum value necessary to maintain the required SHUTDOWN MARGIN.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.4	Verify correct breaker alignment and indicated power available to the SDC pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level < 62.8%;

---NOTE--
The pressurizer water level limit does not apply in MODE 3 until after a bubble has been established in the pressurizer and the pressurizer water level has been lowered to within its normal operating band.
----

b. ≥ 375 kW of pressurizer heater capacity available from electrical bus 1D, and

c. ≥ 375 kW of pressurizer heater capacity available from electrical bus 1E with the capability of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor tripped.	6 hours
	<u>AND</u>	
	A.2 Be in MODE 4.	30 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. < 375 kW pressurizer heater capacity available from electrical bus 1D, or electrical bus 1E, <u>OR</u> Required pressurizer heater capacity from electrical bus 1E not capable of being powered from an emergency power supply.	B.1 Restore required pressurizer heaters to OPERABLE status.	72 hours
C. ------------NOTE--------------- Not applicable when the remaining electrical bus 1D or electrical bus 1E required pressurizer heaters intentionally made inoperable. ----------------------------------- < 375 kW pressurizer heater capacity available from electrical bus 1D, and electrical bus 1E, <u>OR</u> < 375 kW pressurizer heater capacity available from electrical bus 1D, and required pressurizer heater capacity from electrical bus 1E not capable of being powered from an emergency power supply.	C.1 Restore at least electrical bus 1D or electrical bus 1E required pressurizer heaters to OPERABLE status.	24 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	------------------------------NOTE--------------------------- Not required to be met until 1 hour after establishing a bubble in the pressurizer and the pressurizer water level has been lowered to within its normal operating band. --- Verify pressurizer water level is < 62.8%.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify the capacity of pressurizer heaters from electrical bus 1D, and electrical bus 1E is ≥ 375 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.3	Verify the required pressurizer heater capacity from electrical bus 1E is capable of being powered from an emergency power supply.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings as specified in Table 3.4.10-1.

APPLICABILITY: MODES 1 and 2,
MODE 3 with all PCS cold leg temperatures ≥ 430°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Reduce any PCS cold leg temperature < 430°F.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1% of required setpoint.	In accordance with the INSERVICE TESTING PROGRAM

Table 3.4.10-1 (page 1 of 1)
Pressurizer Safety Valve Lift Settings

VALVE NUMBER	LIFT SETTING (psia ± 3%)
RV-1039	2580
RV-1040	2540
RV-1041	2500

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with all PCS cold leg temperatures ≥ 430°F.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each PORV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PORV inoperable.	A.1 Close associated block valve. AND A.2 Restore PORV to OPERABLE status.	1 hour 72 hours
B. One block valve inoperable.	B.1 Place associated PORV in manual control. AND B.2 Restore block valve to OPERABLE status.	1 hour 72 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two PORVs inoperable.	C.1	Close associated block valves.	1 hour
	AND		
	C.2	Restore at least one PORV to OPERABLE status.	2 hours
D. Two block valves inoperable.	D.1	Place associated PORVs in manual control.	1 hour
	AND		
	D.2	Restore at least one block valve to OPERABLE status.	2 hours
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.11.1	Perform a complete cycle of each block valve.	Once prior to entering MODE 4 from MODE 5 if not performed within previous 92 days
SR 3.4.11.2	Perform a complete cycle of each PORV with PCS average temperature > 200°F.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with:

a. Both High Pressure Safety Injection (HPSI) pumps incapable of injecting into the PCS, and

---NOTES--

1. LCO 3.4.12.a is only required when any PCS cold leg temperature is < 300°F.

2. LCO 3.4.12.a does not prohibit the use of the HPSI pumps for emergency addition of makeup to the PCS.

----

b. One of the following pressure relief capabilities:

   1. Two Power Operated Relief Valves (PORVs) with lift settings as specified in Figure 3.4.12-1; or

   2. The PCS depressurized and a PCS vent capable of relieving ≥ 167 gpm at a pressure of 315 psia.

APPLICABILITY: MODE 3 when any PCS cold leg temperature is < 430°F,
MODES 4 and 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable to PORVs when entering MODE 4.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two HPSI pumps capable of injecting into the PCS.	A.1	Initiate action to verify no HPSI pump is capable of injecting into the PCS.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required PORV inoperable and pressurizer water level ≤ 57%.	B.1 Restore required PORV to OPERABLE status.	7 days
C. One required PORV inoperable and pressurizer water level > 57%.	C.1 Restore required PORV to OPERABLE status.	24 hours
D. Two required PORVs inoperable. <u>OR</u> Required Action and associated Completion Time not met. <u>OR</u> LTOP System inoperable for any reason other than Condition A, B, or C.	D.1 Depressurize PCS and establish PCS vent capable of relieving ≥ 167 gpm at a PCS pressure of 315 psia.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	------------------------------NOTE------------------------ Only required to be met when complying with LCO 3.4.12.a. --- Verify both HPSI pumps are incapable of injecting into the PCS.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	Verify required PCS vent, capable of relieving ≥ 167 gpm at a PCS pressure of 315 psia, is open.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.3	Verify PORV block valve is open for each required PORV.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.4	------------------------------NOTE------------------------ Not required to be performed until 12 hours after decreasing any PCS cold leg temperature to < 430°F. --- Perform CHANNEL FUNCTIONAL TEST on each required PORV, excluding actuation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.5	Perform CHANNEL CALIBRATION on each required PORV actuation channel.	In accordance with the Surveillance Frequency Control Program

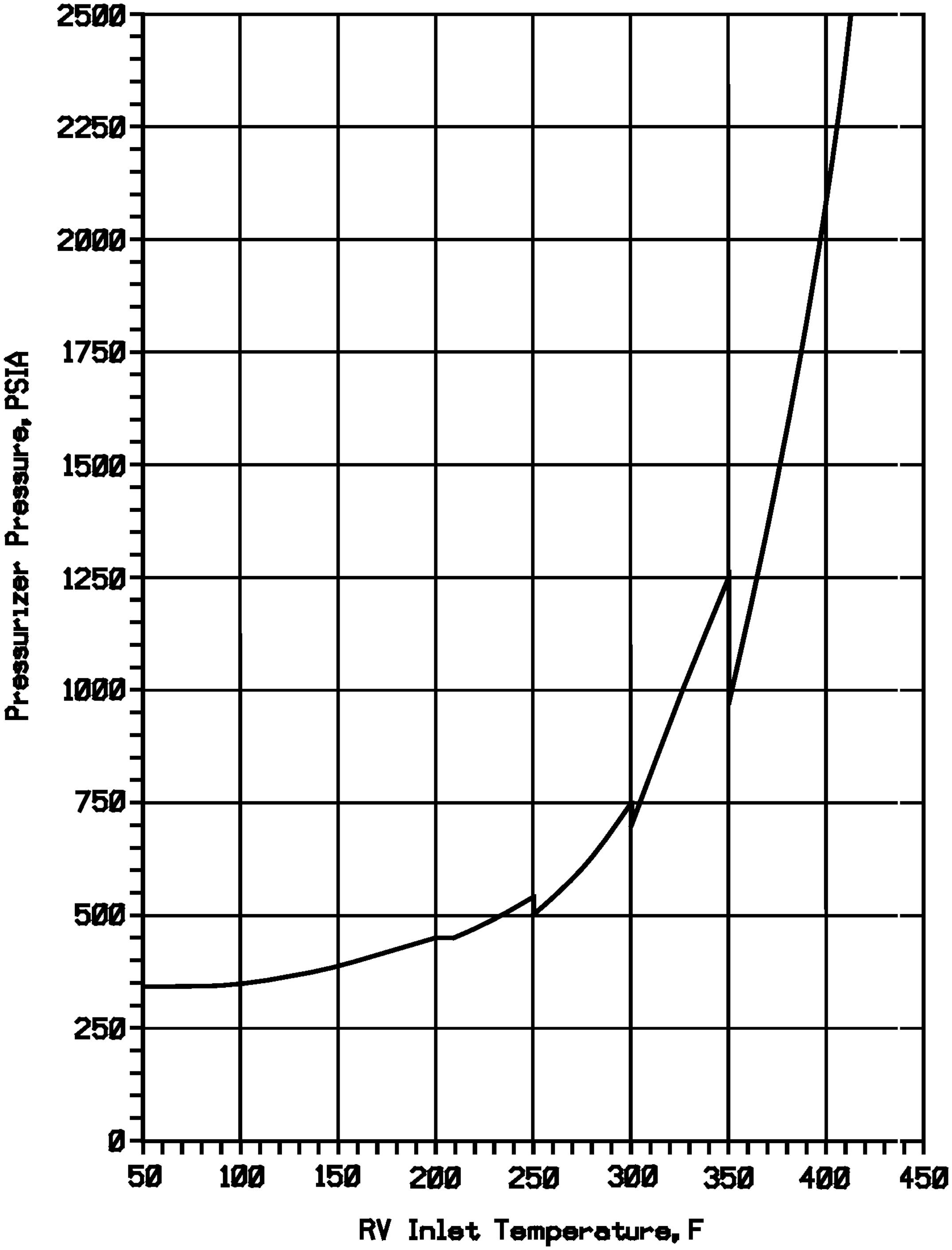


Figure 3.4.12-1 (Page 1 of 1)
LTOP Setpoint Limit
Applicable up to 42.1 EFPY

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.13 PCS Operational LEAKAGE

LCO 3.4.13 PCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. PCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary leakage.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	-------------------------------NOTES------------------------- 1. Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify PCS operational LEAKAGE is within limits by performance of PCS water inventory balance.	----------NOTE-------- Only required to be performed during steady state operation --------------------------- In accordance with the Surveillance Frequency Control Program
SR 3.4.13.2	------------------------------- NOTE --------------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.14 PCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each PCS PIV shall be within limits and both Shutdown Cooling (SDC) suction valve interlocks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except during the SDC mode of operation, or transition to or from, the SDC mode of operation.

ACTIONS

---NOTES---

1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more PCS PIVs not within limit.	------------------NOTE------------------ Each valve used to satisfy Required Action A.1 must have been verified to meet SR 3.4.14.1 and be on the PCS pressure boundary or the high pressure portion of the system. ---	(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	AND		
	A.2	Restore PCS PIV to within limits.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours
C. One or both SDC suction valve interlocks inoperable.	C.1	Isolate the affected penetration by use of one closed deactivated valve.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.14.1	-------------------------------NOTES-------------------------- 1. Only required to be performed in MODES 1 and 2. 2. Leakage rates ≤ 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible leakage rate of 5.0 gpm by 50% or greater. 3. Minimum test differential pressure shall not be less than 150 psid. --- Verify leakage from each PCS PIV is equivalent to ≤ 5 gpm at a PCS pressure of 2060 psia.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once prior to entering MODE 2 whenever the plant has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months
SR 3.4.14.2	Verify each SDC suction valve interlock prevents its associated valve from being opened with a simulated or actual PCS pressure signal ≥ 280 psia.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.3	-------------------------------NOTE---------------------------- Only required to be performed in MODES 1 and 2. --- Verify each of the four Low Pressure Safety Injection (LPSI) check valves are closed.	Prior to entering MODE 2 after each use of the LPSI check valves for SDC

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.15 PCS Leakage Detection Instrumentation

LCO 3.4.15 Three of the following PCS leakage detection instrumentation channels shall be OPERABLE:

a. One containment sump level indicating channel;

b. One containment atmosphere gaseous activity monitoring channel;

c. One containment air cooler condensate level switch channel;

d. One containment atmosphere humidity monitoring channel.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two required leak detection instrument channels inoperable.	A.1	Perform SR 3.4.13.1 (PCS water inventory balance).	Once per 24 hours
	AND		
	A.2	Restore inoperable channel(s) to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. All required channels inoperable.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of the required containment sump level indicator.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.2	Perform CHANNEL CHECK of the required containment atmosphere gaseous activity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.3	Perform CHANNEL CHECK of the required containment atmosphere humidity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.4	Perform CHANNEL FUNCTIONAL TEST of the required containment air cooler condensate level switch.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.5	Perform CHANNEL CALIBRATION of the required containment sump level indicator.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.6	Perform CHANNEL CALIBRATION of the required containment atmosphere gaseous activity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.7	Perform CHANNEL CALIBRATION of the required containment atmosphere humidity monitor.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.16 PCS Specific Activity

LCO 3.4.16 The specific activity of the primary coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with PCS average temperature (T_{ave}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	-----------------NOTE------------------- LCO 3.0.4.c is applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 < 40 μCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 ≥ 40 μCi/gm. <u>OR</u> Gross specific activity of the primary coolant not within limit.	B.1 Be in MODE 3 with T_{ave} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify primary coolant gross specific activity ≤ 100/Ē μCi/gm.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.2	-------------------------------NOTE--------------------------- Only required to be performed in MODE 1. --- Verify primary coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program AND Once between 2 and 6 hours after THERMAL POWER change of ≥ 15% RTP within a 1 hour period
SR 3.4.16.3	-------------------------------NOTE--------------------------- Not required to be performed until 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine Ē from a sample taken in MODE 1 after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

<u>AND</u>

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	<u>AND</u>	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Safety Injection Tanks (SITs)

LCO 3.5.1 Four SITs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SIT inoperable due to boron concentration not within limits. <u>OR</u> One SIT inoperable due to the inability to verify level or pressure.	A.1 Restore SIT to OPERABLE status.	72 hours
B. One SIT inoperable for reasons other than Condition A.	B.1 Restore SIT to OPERABLE status.	24 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
D. Two or more SITs inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each SIT isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	Verify borated water volume in each SIT is $\geq$ 1040 ft^3 and $\leq$ 1176 ft^3.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each SIT is $\geq$ 200 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify boron concentration in each SIT is $\geq$ 1720 ppm and $\leq$ 2500 ppm.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.5	Verify power is removed from each SIT isolation valve operator.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Primary Coolant System (PCS) temperature ≥ 325°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One LPSI subsystem inoperable.	A.1 Restore LPSI subsystem to OPERABLE status.	7 days
B. One or more ECCS trains inoperable for reasons other than Condition A.	B.1 Restore train(s) to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce PCS temperature to < 325°F.	6 hours 24 hours
D. Less than 100% of the required ECCS flow available.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves and hand switches are in the open position. <u>Valve/Hand Switch Number</u> — <u>Function</u> CV-3027 — SIRWT Recirc Valve HS-3027A — Hand Switch For CV-3027 HS-3027B — Hand Switch For CV-3027 CV-3056 — SIRWT Recirc Valve HS-3056A — Hand Switch For CV-3056 HS-3056B — Hand Switch For CV-3056	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.3	Verify CV-3006, "SDC Flow Control Valve," is open and its air supply is isolated.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.2.5	Verify each ECCS automatic valve that is not locked, sealed, or otherwise secured in position, in the flow path actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.7	Verify each LPSI pump stops on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.8	Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. <u>Valve Number</u> — <u>Function</u> MO-3008 — LPSI to Cold leg 1A MO-3010 — LPSI to Cold leg 1B MO-3012 — LPSI to Cold leg 2A MO-3014 — LPSI to Cold leg 2B MO-3082 — HPSI to Hot leg 1 MO-3083 — HPSI to Hot leg 1	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.9	Verify, by visual inspection, the containment sump passive strainer assemblies are not restricted by debris, and the containment sump passive strainer assemblies and other containment sump entrance pathways show no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3 One Low Pressure Safety Injection (LPSI) train shall be OPERABLE.

---NOTE---
A LPSI train may be considered OPERABLE during alignment and operation for shutdown cooling if capable of being manually realigned to the ECCS mode of operation.

APPLICABILITY: MODE 3 with Primary Coolant System (PCS) temperature < 325°F,
MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required LPSI train inoperable.	A.1 Initiate action to restore one LPSI train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	The following SRs of Specification 3.5.2, "ECCS - Operating," are applicable: SR 3.5.2.2 SR 3.5.2.9 SR 3.5.2.4	In accordance with applicable SRs

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Safety Injection Refueling Water Tank (SIRWT)

LCO 3.5.4 The SIRWT shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SIRWT boron concentration not within limits. OR SIRWT borated water temperature not within limits.	A.1 Restore SIRWT to OPERABLE status.	8 hours
B. SIRWT inoperable for reasons other than Condition A.	B.1 Restore SIRWT to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify SIRWT borated water temperature is ≥ 40°F and ≤ 100°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	--------------------------------NOTE-------------------------- Only required to be met in MODES 1, 2, and 3. -- Verify SIRWT borated water volume is ≥ 250,000 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	--------------------------------NOTE-------------------------- Only required to be met in MODE 4. -- Verify SIRWT borated water volume is ≥ 200,000 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.4	Verify SIRWT boron concentration is ≥ 1720 ppm and ≤ 2500 ppm.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Containment Sump Buffering Agent and Weight Requirements

LCO 3.5.5 Buffer baskets shall contain ≥ 8,186 lbs and ≤10,553 lbs of Sodium Tetraborate Decahydrate (STB) $Na_2B_4O_7 \cdot 10H_2O$.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. STB not within limits.	A.1 Restore STB to within limits.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 Be in MODE 4.	30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	Verify the STB baskets contain ≥ 8,186 lbs and ≤10,553 lbs of equivalent weight sodium tetraborate decahydrate.	In accordance with the Surveillance Frequency Control Program
SR 3.5.5.2	Verify that a sample from the STB baskets provides adequate pH adjustment of borated water.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing, except for containment air lock testing, in accordance with the Containment Leak Rate Testing Program.	In accordance with the Containment Leak Rate Testing Program
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Structural Integrity Surveillance Program.	In accordance with the Containment Structural Integrity Surveillance Program

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTES---

1. Entry and exit is permissible through a “locked” air lock door to perform repairs on the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	----------------NOTES---------------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. --	
	A.1 Verify the OPERABLE door is closed in the affected air lock.	1 hour
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed in the affected air lock. <u>AND</u> ------------------NOTE----------------- Air lock doors in high radiation areas may be verified locked closed by administrative means. --- A.3 Verify the OPERABLE door is locked closed in the affected air lock.	24 hours Once per 31 days
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	-----------------NOTES---------------- 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. --- B.1 Verify an OPERABLE door is closed in the affected air lock. <u>AND</u>	 1 hour (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Lock an OPERABLE door closed in the affected air lock. <u>AND</u> ------------------NOTE----------------- Air lock doors in high radiation areas may be verified locked closed by administrative means. --- B.3 Verify an OPERABLE door is locked closed in the affected air lock.	24 hours Once per 31 days
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1. <u>AND</u> C.2 Verify a door is closed in the affected air lock. <u>AND</u> C.3 Restore air lock to OPERABLE status.	Immediately 1 hour 24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	----------------------------NOTES--------------------------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. --- Perform required air lock leakage rate testing in accordance with the Containment Leak Rate Testing Program.	In accordance with the Containment Leak Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTES--

1. Penetration flow paths, except for 8 inch purge exhaust valves and 12 inch air room supply valves penetration flow paths, may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for system(s) made inoperable by containment isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE------------- Only applicable to penetration flow paths with two containment isolation valves. ---------------------------------- One or more penetration flow paths with one containment isolation valve inoperable (except for purge exhaust valve or air room supply valve not locked closed).	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. <u>AND</u>	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	--------------------NOTE---------------- Isolation devices in high radiation areas may be verified by use of administrative means. --- A.2 Verify the affected penetration flow path is isolated.	 Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. --------------NOTE------------- Only applicable to penetration flow paths with two containment isolation valves. ----------------------------------- One or more penetration flow paths with two containment isolation valves inoperable (except for purge exhaust valve or air room supply valve not locked closed).	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. --------------NOTE------------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ---------------------------------- One or more penetration flow paths with one containment isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND -------------------NOTE----------------- Isolation devices in high radiation areas may be verified by use of administrative means. -- C.2 Verify the affected penetration flow path is isolated.	72 hours Once per 31 days
D. One or more purge exhaust or air room supply valves not locked closed.	D.1 Lock closed the affected valves.	1 hour
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 8 inch purge valve and 12 inch air room supply valve is locked closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	-----------------------------NOTE---------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---	
	Verify each manual containment isolation valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured in position, and is required to be closed during accident conditions, is closed, except for containment isolation valves that are open under administrative controls.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	---------------------------NOTE---------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---	
	Verify each manual containment isolation valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured in position, and required to be closed during accident conditions, is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.5	Verify each containment 8 inch purge exhaust and 12 inch air room supply valve is closed by performance of a leakage rate test.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.6	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≤ 1.0 psig in MODES 1 and 2 and ≤ 1.5 psig in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limit.	A.1 Restore containment pressure to within limit.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limit.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 140°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Cooling Systems

LCO 3.6.6 Two containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment cooling trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 30 hours
C. Less than 100% of the required post accident containment cooling capability available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.2	Operate each Containment Air Cooler Fan Unit for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.3	Verify the containment spray piping is full of water to the 735 ft elevation in the containment spray header.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.4	Verify total service water flow rate, when aligned for accident conditions, is ≥ 4800 gpm to Containment Air Coolers VHX-1, VHX-2, and VHX-3.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.5	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.6.6	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to its correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.7	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.8	Verify each containment cooling fan starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.9	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Twenty-three MSSVs shall be OPERABLE as specified in Table 3.7.1-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required MSSVs inoperable.	A.1 Restore required MSSVs to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND	6 hours
	B.2 Be in MODE 4.	30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each required MSSV lift setting is within the limits of Table 3.7.1-1 in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

Table 3.7.1-1 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER		LIFT SETTING (psig ± 3%)
Steam Generator A	Steam Generator B	
RV-0703 RV-0704 RV-0705 RV-0706	RV-0701 RV-0702 RV-0707 RV-0708	1025
RV-0713 RV-0714 RV-0715 RV-0716	RV-0709 RV-0710 RV-0711 RV-0712	1005
RV-0717 RV-0718 RV-0723 RV-0724	RV-0719 RV-0720 RV-0721 RV-0722	985

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when both MSIVs are closed and de-activated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	8 hours
B. Required Action and Associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
C. --------------NOTE------------- Separate Condition entry is allowed for each MSIV. ---------------------------------- One or more MSIVs inoperable in MODE 2 or 3.	C.1 Close MSIV. <u>AND</u> C.2 Verify MSIV is closed.	8 hours Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify closure time of each MSIV is ≤ 5 seconds on an actual or simulated actuation signal from each train under no flow conditions.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

LCO 3.7.3 Two MFRVs and two MFRV bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when both MFRVs and both MFRV bypass valves are either closed and de-activated, or isolated by closed manually actuated valves.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more MFRVs or MFRV bypass valves inoperable.	A.1	Close or isolate inoperable MFRV(s) or MFRV bypass valve(s).	8 hours
	<u>AND</u>		
	A.2	Verify inoperable MFRV(s) or MFRV bypass valve(s) is closed or isolated.	Once per 7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFRV and MFRV bypass valve is ≤ 22 seconds on a actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valves (ADVs)

LCO 3.7.4 One ADV per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is being relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ADV inoperable.	A.1 Restore ADV to OPERABLE status.	7 days
B. Two required ADVs inoperable.	B.1 Restore one ADV to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each ADV.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Two AFW trains shall be OPERABLE.

---NOTES---

1. Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

2. The steam driven pump is only required to be OPERABLE prior to making the reactor critical.

3. Two AFW pumps may be placed in manual for testing, for a period of up to 4 hours.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AFW trains inoperable in MODE 1, 2, or 3.	A.1 Restore train(s) to OPERABLE status.	72 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR Less than 100% of the required AFW flow available to either steam generator. OR Less than two AFW pumps OPERABLE in MODE 1, 2, OR 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 30 hours
C. Less than 100% of the required AFW flow available, to both steam generators.	------------------NOTE------------------ LCO 3.0.3 and all other LCO Required Actions requiring MODE changes or power reductions are suspended until at least 100% of the required AFW flow is available. --- C.1 Initiate action to restore one AFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each required AFW manual, power operated, and automatic valve in each water flow path and in the steam supply flow path to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	------------------------NOTE------------------------------- Not required to be met for the turbine driven AFW pump in MODE 3 below 800 psig in the steam generators. --- Verify the developed head of each required AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.5.3	----------------------------NOTE---------------------------- Only required to be met in MODES 1, 2 or 3 when AFW is not in operation. --- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.4	------------------------------NOTE---------------------------- Only required to be met in MODES 1, 2, and 3. --- Verify each required AFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage and Supply

LCO 3.7.6 The combined useable volume of the Condensate Storage Tank (CST) and Primary Makeup Storage Tank (T-81) shall be ≥ 100,000 gallons.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Condensate volume not within limit.	A.1 Verify OPERABILITY of backup water supplies. <u>AND</u> A.2 Restore condensate volume to within limit.	4 hours <u>AND</u> Once per 12 hours thereafter 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 without reliance on steam generator for heat removal.	6 hours 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify condensate useable volume is ≥ 100,000 gallons.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more CCW trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours
C. Less than 100% of the required post accident CCW cooling capability available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-----------------------------NOTE--------------------------- Isolation of CCW flow to individual components does not render the CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.2	----------------------------NOTE---------------------------- Only required to be met in MODES 1, 2, and 3. --- Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.3	----------------------------NOTE---------------------------- Only required to be met in MODES 1, 2, and 3. --- Verify each CCW pump starts automatically on an actual or simulated actuation signal in the “with standby power available” mode.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SWS trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours
C. Less than 100% of the required post accident SWS cooling capability available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	---------------------------NOTE--------------------------- Isolation of SWS flow to individual components does not render SWS inoperable. -- Verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	---------------------------NOTE--------------------------- Only required to be met in MODES 1, 2, and 3. -- Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	---------------------------NOTE--------------------------- Only required to be met in MODES 1, 2, and 3. -- Verify each SWS pump starts automatically on an actual or simulated actuation signal in the “with standby power available” mode.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. UHS inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of UHS is ≥ 568.25 ft above mean sea level.	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.2	Verify water temperature of UHS is ≤ 85°F.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.10 Control Room Ventilation (CRV) Filtration

LCO 3.7.10 Two CRV Filtration trains shall be OPERABLE.

---------------------------------NOTE---
The control room envelope (CRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4,
During CORE ALTERATIONS,
During movement of irradiated fuel assemblies,
During movement of a fuel cask in or over the Spent Fuel Pool (SFP).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRV Filtration train inoperable for reasons other than Condition B.	A.1 Restore CRV Filtration train to OPERABLE status.	7 days
B. One or more CRV Filtration trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4.	B.1 Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>	
	B.2 Verify mitigating actions ensure CRE occupant radiological exposures will not exceed limits, and CRE occupants are protected from chemical and smoke hazards.	24 hours
	<u>AND</u>	
	B.3 Restore CRE boundary to OPERABLE status.	90 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE------------ Not applicable when second CRV Filtration train intentionally made inoperable. --------------------------------- Two CRV Filtration trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 Initiate action to implement mitigating actions. AND C.2 Verify LCO 3.4.16, "PCS Specific Activity," is met. AND C.3 Restore at least one CRV Filtration train to OPERABLE status.	Immediately 1 hour 24 hour
D. Required Action and associated Completion Time of Condition A not met during CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP.	D.1 Place OPERABLE CRV Filtration train in emergency mode. OR D.2.1 Suspend CORE ALTERATIONS. AND D.2.2 Suspend movement of irradiated fuel assemblies. AND D.2.3 Suspend movement of a fuel cask in or over the SFP.	Immediately Immediately Immediately Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CRV Filtration trains inoperable during CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP. OR One or more CRV Filtration trains inoperable due to an inoperable CRE boundary during CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP.	E.1 Suspend CORE ALTERATIONS. <u>AND</u> E.2 Suspend movement of irradiated fuel assemblies. <u>AND</u> E.3 Suspend movement of a fuel cask in or over the SFP.	Immediately Immediately Immediately
F. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, 3, or 4.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRV Filtration train for ≥ 10 continuous hours with associated heater (VHX-26A or VHX-26B) operating.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Perform required CRV Filtration filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.7.10.3	-----------------------------NOTE----------------------------- Only required to be met in MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies in containment. --- Verify each CRV Filtration train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

3.7 PLANT SYSTEMS

3.7.11 Control Room Ventilation (CRV) Cooling

LCO 3.7.11 Two CRV Cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4,
During CORE ALTERATIONS,
During movement of irradiated fuel assemblies,
During movement of a fuel cask in or over the Spent Fuel Pool (SFP).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRV Cooling train inoperable.	A.1 Restore CRV Cooling train to OPERABLE status.	30 days
B. -------------NOTE-------------- Not applicable when second CRV Cooling train intentionally made inoperable. ----------------------------------- Two CRV Cooling trains inoperable in MODE 1, 2, 3, or 4.	B.1 Restore at least one CRV Cooling train to OPERABLE status.	24 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours
D. Required Action and associated Completion Time of Condition A not met during CORE ALTERATIONS, during movement of irradiated fuel assemblies, or movement of a fuel cask in or over the SFP.	D.1	Place OPERABLE CRV Cooling train in operation.	Immediately
	OR		
	D.2.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	D.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	D.2.3	Suspend movement of a fuel cask in or over the SFP.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two CRV Cooling trains inoperable during CORE ALTERATIONS, during movement of irradiated fuel assemblies, or movement of a fuel cask in or over the SFP.	E.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	E.2	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	E.3	Suspend movement of a fuel cask in or over the SFP.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify each CRV Cooling train has the capability to remove the assumed heat load.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.12 Fuel Handling Area Ventilation System

LCO 3.7.12 The Fuel Handling Area Ventilation System shall be OPERABLE with one fuel handling area exhaust fan aligned to the emergency filter bank and in operation.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel handling building when irradiated fuel assemblies with < 30 days decay time are in the fuel handling building,

During movement of a fuel cask in or over the SFP when irradiated fuel assemblies with < 90 days decay time are in the fuel handling building,

During CORE ALTERATIONS when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open,

During movement of irradiated fuel assemblies in the containment when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel Handling Area Ventilation System not aligned or in operation. OR Fuel Handling Area Ventilation System inoperable.	A.1 Suspend movement of fuel assemblies.	Immediately
	AND	
	A.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.3 Suspend movement of a fuel cask in or over the SFP.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Perform required Fuel Handling Area Ventilation System filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.7.12.2	Verify the flow rate of the Fuel Handling Area Ventilation System, when aligned to the emergency filter bank, is ≥ 5840 cfm and ≤ 8760 cfm.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.13 Engineered Safeguards Room Ventilation (ESRV) Dampers

LCO 3.7.13 Two ESRV Damper trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ESRV Damper trains inoperable.	A.1 Initiate action to isolate associated ESRV Damper train(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify each ESRV Damper train closes on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool (SFP) Water Level

LCO 3.7.14 The SFP water level shall be ≥ 647 ft elevation.

---NOTE---
SFP level may be below the 647 ft elevation to support fuel cask movement, if the displacement of water by the fuel cask when submerged in the SFP, would raise SFP level to ≥ 647 ft elevation.

APPLICABILITY: During movement of irradiated fuel assemblies in the SFP,
During movement of a fuel cask in or over the SFP.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies in SFP.	Immediately
	<u>AND</u>	
	A.2 Suspend movement of fuel cask in or over the SFP.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the SFP water level is ≥ 647 ft elevation.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.15 The SFP boron concentration shall be ≥ 1720 ppm.

APPLICABILITY: When fuel assemblies are stored in the Spent Fuel Pool.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the SFP.	Immediately
	AND	
	A.2 Initiate action to restore SFP boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the SFP boron concentration is within limit.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Pool Storage

LCO 3.7.16 Storage in the spent fuel pool shall be as follows:

a. Each fuel assembly and non-fissile bearing component stored in a Region I Carborundum equipped storage rack shall be within the limitations in Specification 4.3.1.1 and, as applicable, within the requirements of the maximum nominal planar average U-235 enrichment and burnup of Tables 3.7.16-2, 3.7.16-3, 3.7.16-4 or 3.7.16-5,

b. Fuel assemblies in a Region I Metamic equipped storage rack shall be within the limitations in Specification 4.3.1.2, and

c. The combination of maximum nominal planar average U-235 enrichment, burnup, and decay time of each fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly or non-fissile bearing component is stored in the spent fuel pool or the north tilt pit.

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to restore the noncomplying fuel assembly or non-fissile bearing component within requirements.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means each fuel assembly or non-fissile bearing component meets fuel storage requirements.	Prior to storing the fuel assembly or non-fissile bearing component in the spent fuel pool

3.7 PLANT SYSTEMS

3.7.17 Secondary Specific Activity

LCO 3.7.17 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify the specific activity of the secondary coolant is within limit.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two Diesel Generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 (offsite source check) for OPERABLE offsite circuit. <u>AND</u> A.2 Restore offsite circuit to OPERABLE status.	1 hour <u>AND</u> Once per 8 hours thereafter 72 hours <u>AND</u> 10 days from discovery of failure to meet LCO

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One DG inoperable.	B.1	Perform SR 3.8.1.1 (offsite source check) for the OPERABLE offsite circuit(s).	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
	B.2	Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	B.3.1	Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	<u>OR</u>		
	B.3.2	Perform SR 3.8.1.2 (start test) for OPERABLE DG.	24 hours
	<u>AND</u>		
	B.4	Restore DG to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet LCO

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. AND C.2 Restore one offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) 24 hours
D. One offsite circuit inoperable. AND One DG inoperable.	------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," when Condition D is entered with no AC power source to any train. --- D.1 Restore offsite circuit to OPERABLE status. OR D.2 Restore DG to OPERABLE status.	 12 hours 12 hours
E. Two DGs inoperable.	E.1 Restore one DG to OPERABLE status.	2 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 5.	6 hours 36 hours
G. Three or more AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and voltage for each offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	Verify each DG starts from standby conditions and achieves: a. In ≤ 10 seconds, ready-to-load status; and b. Steady state voltage ≥ 2280 V and ≤ 2520 V, and frequency ≥ 59.5 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	-----------------------------NOTES--------------------------- 1. Momentary transients outside the load range do not invalidate this test. 2. This Surveillance shall be conducted on only one DG at a time. 3. This Surveillance shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2. --- Verify each DG is synchronized and loaded, and operates for ≥ 60 minutes: a. For ≥ 15 minutes loaded to greater than or equal to peak accident load; and b. For the remainder of the test at a load ≥ 2300 kW and ≤ 2500 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.4	Verify each day tank contains ≥ 2500 gallons of fuel oil.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.5	Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 68 Hz; b. Within 3 seconds following load rejection, the voltage is ≥ 2280 V and ≤ 2640 V; and c. Within 3 seconds following load rejection, the frequency is ≥ 59.5 Hz and ≤ 61.5 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.6	Verify each DG, operating at a power factor ≤ 0.9, does not trip, and voltage is maintained ≤ 4000 V during and following a load rejection of ≥ 2300 kW and ≤ 2500 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.7	------------------------------NOTE------------------------------ This Surveillance shall not be performed in MODE 1, 2, 3, or 4. --	
	Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected shutdown loads through automatic load sequencer, 3. maintains steady state voltage ≥ 2280 V and ≤ 2520 V, 4. maintains steady state frequency ≥ 59.5 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected loads for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	-----------------------------NOTE----------------------------- Momentary transients outside the load and power factor ranges do not invalidate this test. -- Verify each DG, operating at a power factor ≤ 0.9, operates for ≥ 24 hours: a. For ≥ 100 minutes loaded ≥ its peak accident loading; and b. For the remaining hours of the test loaded ≥ 2300 kW and ≤ 2500 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.9	-----------------------------NOTE----------------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. Verify each DG: a. Synchronizes with offsite power source while supplying its associated 2400 V bus upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	-----------------------------NOTE----------------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -- Verify the time of each sequenced load is within ± 0.3 seconds of design timing for each automatic load sequencer.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.11	-----------------------------NOTE----------------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated safety injection signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected emergency loads through its automatic load sequencer, 3. achieves steady state voltage ≥ 2280 V and ≤ 2520 V, 4. achieves steady state frequency ≥ 59.5 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected loads for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown"; and

b. One Diesel Generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The required offsite circuit inoperable.	-----------------NOTE------------------- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. --	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. The required DG inoperable.	B.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	B.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	B.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	B.4 Initiate action to restore required DG to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	For AC sources required to be OPERABLE, the following SRs of Specification 3.8.1, "AC Sources - Operating" are applicable: SR 3.8.1.1 SR 3.8.1.2 SR 3.8.1.4.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel, Lube Oil, and Starting Air

LCO 3.8.3 For each Diesel Generator (DG):

a. The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits, and

b. Both diesel fuel oil transfer systems shall be OPERABLE.

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel oil inventory less than a 7 day supply and greater than a 6 day supply.	A.1 Restore fuel oil inventory to within limits.	48 hours
B. Stored lube oil inventory less than a 7 day supply and greater than a 6 day supply.	B.1 Restore stored lube oil inventory to within limits.	48 hours
C. Fuel transfer system (P-18A) inoperable.	C.1 Restore fuel transfer system to OPERABLE status.	12 hours
D. Fuel transfer system (P-18B) inoperable.	D.1 Restore fuel transfer system to OPERABLE status.	7 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Both fuel transfer systems inoperable.	E.1 Restore one fuel transfer system to OPERABLE status.	8 hours
F. Fuel oil properties other than viscosity, and water and sediment, not within limits.	F.1 Restore stored fuel oil properties to within limits.	30 days
G. Required Action and associated Completion Time not met. <u>OR</u> Stored diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, or F.	G.1 Declare associated DG(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify the fuel oil storage subsystem contains ≥ a 7 day supply of fuel.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.2	Verify stored lube oil inventory is ≥ a 7 day supply.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Fuel Oil Testing Program.	In accordance with the Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG air start receiver pressure is ≥ 200 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.5	Check for and remove excess accumulated water from the fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.6	Verify the fuel oil transfer system operates to transfer fuel oil from the fuel oil storage tank to each DG day tank and engine mounted tank.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The Left Train and Right Train DC electrical power sources shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required DC electrical power source battery charger inoperable.	A.1	Verify functional cross-connected battery charger is connected supplying power to the affected DC train.	2 hours
	<u>AND</u>		
	A.2	Restore required DC electrical power source battery charger to OPERABLE status.	7 days
B. One required DC electrical power source battery inoperable.	B.1	Verify OPERABLE directly connected and functional cross-connected battery chargers are connected supplying power to the affected DC train.	2 hours
	<u>AND</u>		
	B.2	Restore required DC electrical power source battery to OPERABLE status.	24 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 125 V on float charge.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify battery connection resistance is ≤ 50 μohm for inter-cell connections, ≤ 360 μohm for inter-rack connections, and ≤ 360 μohm for inter-tier connections.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	Inspect battery cells, cell plates, and racks for visual indication of physical damage or abnormal deterioration that could degrade battery performance.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.4	Remove visible terminal corrosion and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.5	Verify battery connection resistance is ≤ 50 μohm for inter-cell connections, ≤ 360 μohm for inter-rack connections, and ≤ 360 μohm for inter-tier connections.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.6	Verify each required battery charger supplies ≥ 180 amps at ≥ 125 V for ≥ 8 hours.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.7	----------------------------NOTES-------------------------- 1. The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.8	------------------------------NOTE------------------------------ This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -- Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	In accordance with the Surveillance Frequency Control Program <u>AND</u> 12 months when battery shows degradation or has reached 85% of the expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 DC electrical power source(s) shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems — Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power sources inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required DC electrical power source(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.3 SR 3.8.4.5 SR 3.8.4.2 SR 3.8.4.4 SR 3.8.4.6.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the Left Train and Right Train batteries shall be within limits.

APPLICABILITY: When associated DC electrical power source(s) are required to be OPERABLE.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Category A or B limits.	A.1 Verify pilot cells electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	AND	
	A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours AND Once per 7 days thereafter
	AND	
	A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more batteries with average electrolyte temperature of the representative cells $< 70°F$. <u>OR</u> One or more batteries with one or more battery cell parameters not within Category C limits.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.2	Verify average electrolyte temperature of representative cells is $\geq 70°F$.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.3	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	In accordance with the Surveillance Frequency Control Program

Table 3.8.6-1 (page 1 of 1)
Battery Surveillance Requirements

PARAMETER	CATEGORY A: NORMAL LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: NORMAL LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[(b)(c)]	≥ 1.205	≥ 1.200 <u>AND</u> Average of connected cells ≥ 1.205	Not more than 0.020 below average connected cells <u>AND</u> Average of all connected cells ≥ 1.195

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits.

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 Four inverters shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One inverter inoperable.	-----------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any Preferred AC bus de-energized. --- A.1 Restore inverter to OPERABLE status.	 24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to Preferred AC buses.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 Inverter(s) shall be OPERABLE to support the onsite Class 1E Preferred AC bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems — Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct inverter voltage, frequency, and alignment to required Preferred AC buses.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 Left Train and Right Train AC, DC, and Preferred AC bus electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystems in one train inoperable.	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
B. One Preferred AC bus inoperable.	B.1 Restore Preferred AC bus to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
C. One or more DC electrical power distribution subsystems in one train inoperable.	C.1 Restore DC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours
E. Two or more inoperable distribution subsystems that result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and Preferred AC bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and Preferred AC bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or Preferred AC bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate actions to restore required AC, DC, and Preferred AC bus electrical power distribution subsystems to OPERABLE status.	Immediately
	AND	
	A.2.5 Declare associated required shutdown cooling train inoperable and not in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments and voltage to required AC, DC, and Preferred AC bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Primary Coolant System and the refueling cavity shall be maintained at the REFUELING BORON CONCENTRATION.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS. <u>AND</u> A.2 Suspend positive reactivity additions. <u>AND</u> A.3 Initiate action to restore boron concentration to within limit.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is at the REFUELING BORON CONCENTRATION.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range channels shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One source range channel inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
B. Two source range channels inoperable.	B.1 Initiate action to restore one source range channel to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1 (PCS boron concentration verification).	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.9.2.2	-----------------------------NOTE------------------------------ Neutron detectors are excluded from the CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

---NOTE---
The equipment hatch is only required to be closed when the Fuel Handling Area Ventilation System is not in compliance with LCO 3.7.12, "Fuel Handling Area Ventilation System."
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b. One door in the personnel air lock closed;

---NOTE---
One door in the personnel air lock is only required to be closed when the equipment hatch is closed.
----

c. One door in the emergency air lock closed; and

d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Refueling Containment High Radiation Initiation signal.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required to be met containment penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
SR 3.9.3.2	----------------------------NOTE---------------------------- Only required to be met for unisolated containment penetrations. --- Verify each required automatic isolation valve closes on an actual or simulated Refueling Containment High Radiation signal.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

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

LCO 3.9.4 One SDC train shall be OPERABLE and in operation.

---NOTES---

1. The required SDC train may not be in operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Primary Coolant System boron concentration.

2. The required SDC train may be made inoperable for ≤ 2 hours per 8 hour period for testing or maintenance, provided one SDC train is in operation providing flow through the reactor core, and core outlet temperature is ≤ 200°F.

APPLICABILITY: MODE 6 with the refueling cavity water level ≥ 647 ft elevation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required SDC train inoperable or not in operation.	A.1 Initiate action to restore SDC train to OPERABLE status and operation.	Immediately
	<u>AND</u>	
	A.2 Suspend operations involving a reduction in primary coolant boron concentration.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>		
	A.4	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one SDC train is in operation and circulating primary coolant at a flow rate of ≥ 1000 gpm.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

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

LCO 3.9.5 Two SDC trains shall be OPERABLE, and one SDC train shall be in operation.

APPLICABILITY: MODE 6 with the refueling cavity water level < 647 ft elevation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SDC train inoperable.	A.1 Initiate action to restore SDC train to OPERABLE status. OR A.2 Initiate action to establish the refueling cavity water level ≥ 647 ft elevation.	Immediately Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No SDC train OPERABLE or in operation.	B.1 Suspend operations involving a reduction in primary coolant boron concentration.	Immediately
	AND	
	B.2 Initiate action to restore one SDC train to OPERABLE status and to operation.	Immediately
	AND	
	B.3 Initiate action to close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one SDC train is in operation and circulating primary coolant at a flow rate of ≥ 1000 gpm.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.6 Refueling Cavity Water Level

LCO 3.9.6 The refueling cavity water level shall be maintained ≥ 647 ft elevation.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling cavity water level is ≥ 647 ft elevation.	In accordance with the Surveillance Frequency Control Program

4.0 DESIGN FEATURES

4.1 Site Location

The Palisades Nuclear Plant is located on property owned by Holtec Palisades, LLC on the eastern shore of Lake Michigan approximately four and one-half miles south of the southern city limits of South Haven, Michigan. The minimum distance to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 677 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor core shall contain 204 fuel assemblies. Each assembly shall consist of a matrix of zircaloy-4 or M5 clad fuel rods with an initial composition of depleted, natural, or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution. Poison may be placed in the fuel bundles for long-term reactivity control.

4.2.2 Control Rod Assemblies

The reactor core shall contain 45 control rods. Four of these control rods may consist of part-length absorbers. The control material shall be silver-indium-cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The Region I (See Figure B 3.7.16-1) Carborundum equipped fuel storage racks incorporating Regions 1A, 1B, 1C, 1D, and 1E are designed and shall be maintained with:

a. New or irradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.54 weight percent;

4.3.1 Criticality (continued)

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

d. Regions 1A, 1B, and 1C have a nominal 10.25 inch center to center distance between fuel assemblies;

e. Regions 1D and 1E have a nominal 11.25 inch by 10.69 inch center to center distance between fuel assemblies;

f. Region 1A is defined as a subregion of the Region I storage racks located in the main spent fuel pool and is subject to the following restrictions. Fuel assemblies (or fissile bearing components) located in Region 1A shall be in a maximum of two-of-four checkerboard loading pattern of two fuel assemblies (or fissile bearing components) and two empty cells. Designated empty cells may contain non-fuel bearing components in accordance with Section 4.3.1.1m.2. below;

g. Region 1B is defined as a subregion of the Region I storage racks located in the main spent fuel pool and is subject to the following restrictions. Fuel assemblies (or fissile bearing components) located in Region 1B shall be in a maximum of three-of-four loading pattern consisting of three fuel assemblies (or fissile bearing components) and one empty cell. Fuel assemblies in Region 1B shall meet the enrichment dependent burnup restrictions listed in Table 3.7.16-2. Designated empty cells may contain non-fuel bearing components in accordance with Section 4.3.1.1m.2. below;

h. Region 1C is defined as a subregion of the Region I storage racks located in the main spent fuel pool and is subject to the following restrictions. Fuel assemblies (or fissile bearing components) located in Region 1C may be in a maximum of four-of-four loading pattern with no required empty cells. Fuel assemblies in Region 1C shall meet the enrichment dependent burnup restrictions listed in Table 3.7.16-3;

i. Interface requirements for the main spent fuel pool between Region 1A, 1B, and 1C are as follows. Region 1A, 1B, and 1C can be distributed in Region I, in the main spent fuel pool, in any manner provided that any two-by-two grouping of storage cells and the assemblies in them correspond to the requirements of 4.3.1.1f., 4.3.1.1g., or 4.3.1.1h. above;

4.3 Fuel Storage

4.3.1 Criticality (continued)

j. Region 1D is defined as a subregion of the Region I storage rack located in the north tilt pit and is subject to the following restrictions. Fuel assemblies (or fissile bearing components) located in Region 1D may be in a maximum of three-of-four loading pattern consisting of three fuel assemblies (or fissile bearing components) and one empty cell. Fuel assemblies in Region 1D shall meet the enrichment dependent burnup restrictions listed in Table 3.7.16-4;

k. Region 1E is defined as a subregion of the Region I storage rack located in the north tilt pit and is subject to the following restrictions. Fuel assemblies (or fissile bearing components) located in Region 1E may be in a maximum of four-of-four loading pattern with no required empty cells. Fuel assemblies in Region 1E shall meet the enrichment dependent burnup restrictions listed in Table 3.7.16-5;

l. Interface requirements for the north tilt pit between Region 1D and 1E are as follows. Region 1D and 1E can be distributed in Region I in the north tilt pit in any manner provided that any two-by-two grouping of storage cells and the assemblies in them correspond to the requirements of 4.3.1.1j. or 4.3.1.1k. above;

m. Non-fissile bearing component restrictions are as follows:

1. Non-fissile material components may be stored in any designated fuel location in Region 1A, 1B, 1C, 1D, or 1E without restriction.

2. The following non-fuel bearing components (NFBC) may be stored face adjacent to fuel in any designated empty cell in Region 1A or 1B.

(i) The gauge dummy assembly and the lead dummy assembly may be stored face adjacent to fuel in any designated empty cells with no minimum required separation distance.

(ii) A component comprised primarily of stainless steel that displaces less than 30 square inches of water in any plane within the active fuel region may be stored in any designated empty cell as long as the NFBC is at least ten locations away from any other NFBC that is in a designated empty cell, with the exception of 4.3.1.1m.2.(i) above.

4.3.1 Criticality (continued)

3. Control blades may be stored in both fueled and unfueled locations in Regions 1D and 1E, with no limitation on the number.

4.3.1.2 The Region I (See Figure B 3.7.16-1) Metamic equipped fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

d. A nominal 10.25 inch center to center distance between fuel assemblies;

e. New or irradiated fuel assemblies;

f. Two empty rows of storage locations shall exist between the fuel assemblies in a Carborundum equipped rack and the fuel assemblies in an adjacent Metamic equipped rack; and

g. A minimum Metamic B^{10} areal density of 0.02944 g/cm^2.

4.3.1.3 The Region II fuel storage racks (See Figure B 3.7.16-1) are designed and shall be maintained with:

a. Fuel assemblies having maximum nominal planar average U-235 enrichment of 4.60 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR.

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowance for uncertainties as described in Section 9.11 of the FSAR.

d. A nominal 9.17 inch center to center distance between fuel assemblies; and

4.3 Fuel Storage

4.3.1 Criticality (continued)

e. New or irradiated fuel assemblies which meet the maximum nominal planar average U-235 enrichment, burnup, and decay time requirements of Table 3.7.16-1

4.3.1.4 The new fuel storage racks are designed and shall be maintained with:

a. Twenty four unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1, or

Thirty six unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.05 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1;

b. Keff ≤ 0.95 when flooded with either full density or low density (optimum moderation) water including allowances for uncertainties as described in Section 9.11 of the FSAR.

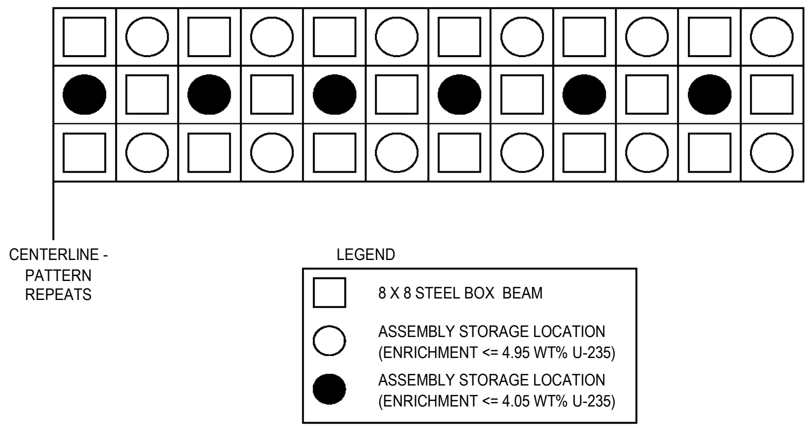
c. The pitch of the new fuel storage rack lattice being ≥ 9.375 inches and every other position in the lattice being permanently occupied by an 8" x 8" structural steel or core plugs, resulting in a nominal 13.26 inch center to center distance between fuel assemblies placed in alternating storage locations.

4.3.2 Drainage

The spent fuel storage pool cooling system suction and discharge piping is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 644 ft 5 inches.

4.3.3 Capacity

The spent fuel storage pool and north tilt pit are designed and shall be maintained with a storage capacity limited to no more than 892 fuel assemblies.



Note: If any assemblies containing fuel enrichments greater than 4.05% U-235 are stored in the New Fuel Storage Rack, the center row must remain empty.

Figure 4.3-1 (page 1 of 1)
New Fuel Storage Rack Arrangement

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage to the engineered safeguards rooms, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident, to as low as practical. The systems include the Containment Spray System, the Safety Injection System, the Shutdown Cooling System, and the containment sump suction piping. This program shall include the following:

a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

c. The portion of the shutdown cooling system that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 255 psig.

d. Piping from valves CV-3029 and CV-3030 to the discharge of the safety injection pumps and containment spray pumps shall be hydrostatically tested at no less than 100 psig.

e. The maximum allowable leakage from the recirculation heat removal systems' components (which include valve stems, flanges and pump seals) shall not exceed 0.2 gallon per minute under the normal hydrostatic head from the SIRW tank.

5.5.3 (Deleted)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the Offsite Dose Calculation Manual (ODCM), (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402.

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

d. Limitation on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each plant to unrestricted areas conforming to 10 CFR 50, Appendix I,

e. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

   2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

f. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary conforming to 10 CFR 50, Appendix I,

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

g. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each plant to areas beyond the site boundary conforming to 10 CFR 50, Appendix I,

h. Limitations on the annual doses or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Containment Structural Integrity Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Containment Structural Integrity Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE and IWL.

If, as a result of a tendon inspection, corrective retensioning of five percent (8) or more of the total number of dome tendons is necessary to restore their liftoff forces to within the limits, a dome delamination inspection shall be performed within 90 days following such corrective retensioning. The results of this inspection shall be reported to the NRC in accordance with Specification 5.6.7, “Containment Structural Integrity Surveillance Report.”

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Containment Structural Integrity Surveillance Program inspection frequencies.

5.5.6 Primary Coolant Pump Flywheel Surveillance Program

a. Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the upper flywheels each 10 years.

b. The provisions of SR 3.0.2 are not applicable to the Flywheel Testing Program

5.5 Programs and Manuals

5.5.7 (Deleted)

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

    1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

b. Performance criteria for SG tube integrity. (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.3 gpm.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "PCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged. The following alternative repair criteria shall be applied as an alternate to the 40% depth based criteria:

1. Tubes found by inservice inspection to contain service induced flaws within 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, shall be plugged. Flaws located below this elevation may remain in service.

2. Tubes found by inservice inspection to contain service induced flaws within 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, shall be plugged. Flaws located below this elevation may remain in service.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, and that may satisfy the applicable tube repair criteria. The tube to tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and

5.5.8 Steam Generator (SG) Program

d. Provisions for SG tube inspections. (continued)

location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

3. If crack indications are found in any SG tube from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

4. When the alternate repair criteria of TS 5.5.8c.1 are implemented, inspect 100% of the inservice tubes to the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 5.5.8c.1 every 24 effective full power months, or one refueling outage, whichever is less.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

5.5.9 Secondary Water Chemistry Program

A program shall be established, implemented and maintained for monitoring of secondary water chemistry to inhibit steam generator tube degradation and shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical variables,

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

d. Procedures for the recording and management of data,

e. Procedures defining corrective actions for all off-control point chemistry conditions, and

f. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

5.5.10 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of Control Room Ventilation (CRV) and Fuel Handling Area Ventilation (FHAV) systems at the frequencies specified in Regulatory Guide 1.52, Revision 2 (RG 1.52), and in accordance with RG 1.52 and ASME N510-1989, at the system flowrates and tolerances specified below*:

a. Demonstrate for each of the ventilation systems that an inplace test of the High Efficiency Particulate Air (HEPA) filters shows a penetration and system bypass < 0.05% for the CRV system and < 1.00% for the FHAV system when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Flowrate (CFM)
FHAV (single fan operation)	7300 ± 20%
FHAV (dual fan operation)	10,000 ± 20%
CRV	3,200 +10% -5%

5.5.10 Ventilation Filter Testing Program (Continued)

b. Demonstrate for each of the ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% for the CRV system and < 1.00% for the FHAV system when tested in accordance with RG 1.52 and ASME N510-1989.

Ventilation System	Flowrate (CFM)
FHAV (dual fan operation)	10,000 ± 20%
CRV	3200 +10% -5%

c. Demonstrate for each of the ventilation systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in RG 1.52 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and equal to the relative humidity specified as follows:

Ventilation System	Penetration	Relative Humidity
FHAV	6.00%	95%
CRV	0.157%	70%

d. For each of the ventilation systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Delta P (In H_20)	Flowrate (CFM)
FHAV (dual fan operation)	6.0	10,000 ± 20%
CRV	8.0	3200 +10% -5%

e. Demonstrate that the heaters for the CRV system dissipates the following specified value ± 20% when tested in accordance with ASME N510-1989:

Ventilation System	Wattage
CRV	15 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Ventilation Filter Testing Program frequencies.

* Should the 720-hour limitation on charcoal adsorber operation occur during a plant operation requiring the use of the charcoal adsorber - such as refueling - testing may be delayed until the completion of the plant operation or up to 1,500 hours of filter operation; whichever occurs first.

5.5 Programs and Manuals

5.5.11 Fuel Oil Testing Program

A fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling requirements, testing requirements, and acceptance criteria, based on the diesel manufacturer's specifications and applicable ASTM Standards. The program shall establish the following:

a. Acceptability of new fuel oil prior to addition to the Fuel Oil Storage Tank, and acceptability of fuel oil stored in the Fuel Oil Storage Tank, by determining that the fuel oil has the following properties within limits:

   1. API gravity or an absolute specific gravity,

   2. Kinematic viscosity, and

   3. Water and sediment content.

b. Other properties of fuel oil stored in the Fuel Oil Storage Tank, specified by the diesel manufacturers or specified for grade 2D fuel oil in ASTM D 975, are within limits.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Fuel Oil Testing Program.

5.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5 Programs and Manuals

5.5.12 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.12.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.13 Safety Functions Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

5.5.13 Safety Functions Determination Program (SFDP) (Continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 Containment Leak Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 2-A, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated October 2008, with the following exceptions:

   1. Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

      Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

   2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at $\geq$10 psig instead.

   3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 54.2 psig. The containment design pressure is 55 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

5.5.14 Containment Leak Rate Testing Program (Continued)

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criteria is $\leq$ 1.0 L_a. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and Type C tests and $\leq$ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage is $\leq$ 1.0 L_a when tested at $\geq$ P_a and combined with all penetrations and valves subjected to Type B and C tests. However, during the first unit startup following testing performed in accordance with this program, the leakage rate acceptance criteria is < 0.6 L_a when combined with all penetrations and valves subjected to Type B and C tests.

b) For each Personnel Air Lock door, leakage is $\leq$ 0.023 L_a when pressurized to $\geq$ 10 psig.

c) For each Emergency Escape Air Lock door, a seal contact check , consisting of a verification of continuous contact between the seals and the sealing surfaces, is acceptable.

di. “Containment OPERABILITY” is equivalent to "Containment Integrity" for the purposes of the testing requirements.

dii. The provisions of SR 3.0.3 are applicable to the Containment Leak Rate Testing Program requirements.

diii. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5 Programs and Manuals

5.5.15 Process Control Program

a. The Process Control Program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 10 CFR 71, Federal and State regulations, and other requirements governing the disposal of the radioactive waste.

b. Changes to the Process Control Program:

1. Shall be documented and records of reviews performed shall be retained as required by the Quality Program. This documentation shall contain:

a) Sufficient information to support the change together with the appropriate analyses or evaluation justifying the change(s) and

b) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after approval by the plant manager.

5.5.16 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation (CRV) Filtration, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRV Filtration, operating at the flow rate required by the Ventilation Filter Testing Program, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5 Programs and Manuals

5.5.17 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 (Deleted)

5.6.2 Radiological Environmental Operating Report

The Radiological Environmental Operating Report covering the operation of the plant during the previous calendar year shall be submitted before May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering operation of the plant in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual and Process Control Program, and shall be in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 (Deleted)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1 Shutdown Margin
3.1.6 Regulating Rod Group Position Limits
3.2.1 Linear Heat Rate Limits
3.2.2 Radial Peaking Factor Limits
3.2.4 ASI Limits
3.4.1 DNB Limits

5.6 Reporting Requirements

5.6.5 COLR (Continued)

b. The analytical methods used to determine the core operating limits shall be those approved by the NRC, specifically those described in the latest approved revision of the following documents:

1. EMF-96-029(P)(A) Volumes 1 and 2, “Reactor Analysis System for PWRs,” Siemens Power Corporation. (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

2. ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation. (Bases report not approved) (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

4. EMF-84-093(P)(A), “Steam Line Break Methodology for PWRs, “Siemens Power Corporation. (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

5. XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company. (Bases document not approved) (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

6. EMF-2310 (P)(A), Revision 0, Framatome ANP, Inc., May 2001, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors.” (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

7. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, & 3.2.2)

8. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

9. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation. (LCOs 3.2.1, 3.2.2, & 3.2.4)

5.6 Reporting Requirements

5.6.5 COLR (Continued)

10. XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

11. XN-NF-82-06(P)(A) and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

12. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWD/MTU," Advanced Nuclear Fuels Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

13. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

14. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

15. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation. (LCOs 3.1.6, 3.2.1, & 3.2.2)

16. ANF-87-150 Volume 2, "Palisades Modified Reactor Protection System Report: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation. [Approved for use in the Palisades design during the NRC review of license Amendment 118, November 15, 1988] (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.4.1)

17. EMF-1961(P)(A), Revision 0, Siemens Power Corporation, July 2000, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors." (LCOs 3.1.6, 3.2.1, 3.2.2, 3.2.4, & 3.4.1)

18. EMF-2328 (P)(A), Revision 0, Framatome ANP, Inc., March 2001, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." (LCOs 3.1.6, 3.2.1, & 3.2.2)

19. BAW-2489P, "Revised Fuel Assembly Growth Correlation for Palisades." (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

20. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors." (LCOs 3.1.6, 3.2.1, & 3.2.2)

5.6 Reporting Requirements

5.6.5 COLR (Continued)

21. BAW-10240(P)-A, “Incorporation of M5 Properties in Framatome ANP Approved Methods.” (LCOs 3.1.6, 3.2.1, 3.2.2, 3.2.4, & 3.4.1)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC.

5.6.6 Post Accident Monitoring Report

When a report is required by LCO 3.3.7, "Post Accident Monitoring Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to OPERABLE status.

5.6.7 Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Dome Delamination tests within 90 days after completion of the tests.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

5.6 Reporting Requirements

5.6.8 Steam Generator Tube Inspection Report (continued)

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

i The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP), or equivalent, that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP, or equivalent, while performing their assigned duties, provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area, and who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP, or equivalent, while performing radiation surveys in such areas, provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, and with the means to communicate with and control every individual in the area, or

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; and who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the “As Low As is Reasonably Achievable” principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

PALISADES PLANT

RENEWED FACILITY OPERATING LICENSE DPR-20

APPENDIX B

ENVIRONMENTAL PROTECTION PLAN
(NON-RADIOLOGICAL)

PALISADES PLANT

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

TABLE OF CONTENTS

1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of environmental values during construction and operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the plant is operated in an environmentally acceptable manner, as established by the FES and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES which relate to water quality matters are regulated by way of the licensee's NPDES permit.

2.0 Environmental Protection Issues

In the final addendum to the FES-OL dated February 1978 the staff considered the environmental impacts associated with the operation of the Palisades Plant. Certain environmental issues were identified which required study or license conditions to resolve environmental concerns and to assure adequate protection of the environment.

2.1 Aquatic Issues

Specific aquatic issues raised by the staff in the FES-OL were:

The need for aquatic monitoring programs to confirm that thermal mixing occurs as predicted, that chlorine releases are controlled within those discharge concentrations evaluated, and that effects on aquatic biota and water quality due to plant operation are no greater than predicted.

Aquatic issues are addressed by the effluent limitations, and monitoring requirements are contained in the effective NPDES permit issued by the State of Michigan, Department of Natural Resources. The NRC will rely on this agency for regulation of matters involving water quality and aquatic biota.

2.2 Terrestrial Issues

1. Potential impacts on the terrestrial environment associated with drift from the mechanical draft cooling towers. (FES-OL addendum Section 6.3)

3.0 Consistency Requirements

3.1 Plant Design and Operation

The licensee may make changes in station design or operation or perform tests or experiments affecting the environment provided such changes, tests or experiments do not involve an unreviewed environmental question, and do not involve a change in the Environmental Protection Plan. Changes in plant design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this section.

Before engaging in additional construction or operational activities which may affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation. When such activity involves a change in the Environmental Protection Plan, such activity and change to the Environmental Protection Plan may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the final environmental statement (FES) as modified by staff's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level [in accordance with 10 CFR Part 51.5(b)(2)] or (3) a matter not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

The licensee shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include a written evaluation which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question nor constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0. The licensee shall include as part of his Annual Environmental Operating Report (per Subsection 5.4.1) brief descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

3.2 Reporting Related to the NPDES Permits and State Certifications

Violations of the NPDES Permit or the State certification (pursuant to Section 401 of the Clean Mater Act) shall be reported to the NRC by submittal of copies of the reports required by the NPDES Permit or certification.

Changes and additions to the NPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The NRC shall be notified of changes to the effective NPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The notification of a licensee-initiated change shall include a copy of the requested revision submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NPDES permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, or local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and promptly reported to the NRC within 24 hours by telephone, telegraph, or facsimile transmissions followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events, onsite plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, fish kills, increase in nuisance organisms or conditions and unanticipated or emergency discharge of waste water or chemical substances.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Meteorological Monitoring

A meteorological monitoring program shall be conducted in the vicinity of the plant site for at least two years after conversion to cooling towers to document effects of cooling tower operation on meteorological variables. Data on the following meteorological variables shall be obtained from the station network shown in Figure 4.2.1: precipitation, temperature, humidity, solar radiation, downcoming radiation, visibility, wind direction and wind speed. In addition, studies shall be conducted for at least two years to measure affects of cooling tower drift on vegetation by associated salt deposition, icing or other causes.

5.0 Administrative Procedures

5.1 Review and Audit

The licensee shall provide for review and audit of compliance with the Environmental Protection Plan. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of plant operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to plant structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the plant. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.3 Changes in Environmental Protection Plan

Request for change in the Environmental Protection Plan shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the Environmental Protection Plan.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following issuance of the operating license. The period of the first report shall begin with the date of issuance of the operating license.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 of this Environmental Protection Plan for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous nonradiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends towards irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of action to alleviate the problem.

The Annual Environmental Operating Report shall also include:

(a) A list of EPP noncompliances and the corrective actions taken to remedy them.

(b) A list of all changes in station design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental issue. |

(c) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of nonroutine event. The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided a copy of such report at the time it is submitted to the other agency.

Enclosure Attachment 3 to

HDI PNP 2023-030

Proposed Technical Specifications Bases Changes

(for information only)

586 pages follow

INSERT Bases 2.1.1

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

The Palisades Nuclear Plant design criteria (Ref. 1) requires, and these SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and Anticipated Operational Occurrences (AOOs). This is accomplished by having a Departure from Nucleate Boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the primary coolant. Overheating of the fuel is prevented by maintaining the steady state, peak Linear Heat Rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the primary coolant.

Operation above the boundary of the nucleate boiling regime beyond onset of DNB could result in excessive cladding temperature because of the resultant sharp reduction in the heat transfer coefficient in the transition and film boiling regimes. If a steam film is allowed to form, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the primary coolant.

BASES

BACKGROUND (continued)

The Reactor Protective System (RPS), in combination with the LCOs, is designed to prevent any anticipated combination of transient conditions for Primary Coolant System (PCS) temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

Palisades uses three DNB correlations; the XNB, ANFP, and HTP detailed in References 3 through 8. The DNB correlations are used solely as analytical tools to ensure that plant conditions will not degrade to the point where DNB could be challenged. The XNB correlation is used for non-High Thermal Performance (HTP) assemblies (assemblies loaded prior to cycle 9), when the non-HTP assemblies could have been limiting. The XNB correlation provides administrative justification for using non-HTP assemblies in Palisades low leakage core design. The ANFP and HTP correlations are used for Palisades High Thermal Performance (HTP) fuel assemblies (assemblies loaded in cycle 9 and later).

The HTP correlation can be used when the calculated reactor coolant conditions fall within the correlation's applicable coolant condition ranges. Outside of the applicable range of the HTP correlation, the ANFP correlation can be used. The ANFP correlation may be used over a broader range of coolant conditions than the HTP correlation. The HTP correlation is an extension of the ANFP correlation and incorporates the results of test sections designed to represent HTP fuel design for CE plants.

INSERT Bases 2.1.1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The prediction of DNB is a function of several measured parameters. The following trip functions and LCOs, limit these measured parameters to protect the Palisades reactor from approaching conditions that could lead to DNB:

Parameter	Protection
Core Flow Rate	Low PCS Flow Trip
Core Power	Variable High Power Trip
PCS Pressure/Core Power	TM/LP Trip
Core Inlet Temperature	T_{inlet} LCO
Axial Shape Index (ASI)	ASI LCO
Assembly Power	Incore Power Monitoring (LHR and F_R^T LCOs)

The RPS setpoints, LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for PCS temperature, pressure, and THERMAL POWER level that would result in a Departure from Nucleate Boiling Ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

The SL represents a design requirement for establishing the protection system trip setpoints identified previously. LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.2, “TOTAL RADIAL PEAKING FACTOR (F_R^T)," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

SL 2.1.1.1 and SL 2.1.1.2 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature remains below melting.

The minimum value of the DNBR during normal operation and design basis AOOs is limited to the following DNB correlation safety limit:

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

The fuel centerline melt LHR value assumed in the safety analysis is 21 kw/ft. Operation ≤ 21 kw/ft maintains the dynamically adjusted peak LHR and ensures that fuel centerline melt will not occur during normal operating conditions or design AOOs.

INSERT Bases 2.1.1

BASES

APPLICABILITY

SL 2.1.1.1 and SL 2.1.1.2 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions are available to prevent PCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the plant into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

In MODES 3, 4, 5, and 6, a reactor core SL is not required, since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the plant in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE where this SL is not applicable and reduces the probability of fuel damage.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Chapter 14
3. XN-NF-621(A), Rev 1
4. XN-NF-709
5. ANF-1224(A), May 1989
6. ANF-89-192, January 1990
7. XN-NF-82-21, Rev 1
8. EMF-92-153(A) and Supplement 1, March 1994

INSERT Bases 2.1.2

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Primary Coolant System (PCS) Pressure SL

BASES

BACKGROUND

The SL on PCS pressure protects the integrity of the PCS against overpressurization. In the event of fuel cladding failure, fission products are released into the primary coolant. The PCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on PCS pressure, continued PCS integrity is ensured. According to Palisades Nuclear Plant design criteria (Ref. 1), the Primary Coolant Pressure Boundary (PCPB) design conditions are not to be exceeded during normal operation and Anticipated Operational Occurrences (AOOs). Also, according to Palisades Nuclear Plant design criteria (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the PCPB greater than limited local yielding.

The design pressure of the PCS is 2500 psia. During normal operation and AOOs, the PCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2) and by the piping, valve, and fitting limit of 120% of design pressure (Ref. 6). The initial hydrostatic test was conducted at 125% of design pressure (3125 psia) to verify the integrity of the primary coolant system (Ref. 2). Following inception of plant operation PCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the PCS could result in a breach of the PCPB. If this occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to applicable limits on radioactive releases specified in 10 CFR 100 and 10 CFR 50.67 (Refs. 4 and 7).

APPLICABLE SAFETY ANALYSES

The PCS primary safety valves, the Main Steam Safety Valves (MSSVs), and the High Pressurizer Pressure trip have settings established to ensure that the PCS pressure SL will not be exceeded.

The PCS primary safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence the valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Protective System (RPS) trip setpoints (LCO 3.3.1, “Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the primary safety valves, provide pressure protection for normal operation and AOOs. In particular, the High Pressurizer Pressure Trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Conservative values for all system parameters, delay times and core moderator coefficient are assumed.

More specifically, for the limiting case, no credit is taken for operation of any other pressure relieving system including the following:

a. Pressurizer Power Operated Relief Valves (PORVs);

b. Turbine Bypass Control System;

c. Atmospheric Steam Dump Valves;

d. Pressurizer Level Control System; or

e. Pressurizer Pressure Control System.

SAFETY LIMITS

The maximum transient pressure allowable in the PCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the PCS piping, valves, and fittings under 120% of design pressure (Ref. 6). The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable PCS pressure is established at 2750 psia.

INSERT Bases 2.1.2

BASES

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, 5, and 6 because this SL could be approached or exceeded in these MODES due to overpressurization events. In MODE 6 with the reactor vessel head installed and the reactor vessel head closure bolts less than fully tensioned the potential for an over pressurization event still exists. Although overpressurization of the PCS is impossible once the reactor vessel head is removed, the requirements of this SL apply as long as fuel is in the reactor. Once all the fuel has been removed from the reactor, the requirements of SL 2.1.2 no longer apply.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the PCS pressure SLs.

2.2.2.1

If the PCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

With PCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater than the value specified in SL 2.1.2 exceeds 110% of the PCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce PCS pressure by terminating the cause of the pressure increase, removing mass or energy from the PCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

If the PCS pressure SL is exceeded in MODE 3, 4, 5 or 6, PCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the PCS pressure SL in MODE 3, 4, 5 or 6 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

INSERT Bases 2.1.2

BASES

REFERENCES

1. FSAR, Section 5.1
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000
4. 10 CFR 100
5. FSAR, Section 4.3
6. ASA B31.1-1955, Code for Pressure Piping, 1967
7. 10 CFR 50.67

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

through

9

BASES

LCO	LCO 3.0.1 ~~and~~ LCO 3.0.2 establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the ~~facility~~ is in the specified conditions of the Applicability statement of each Specification).

MODES or other

plant

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

, unless otherwise specified

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

LCO 3.0.2 INSERT

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

~~Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.~~

LCO 3.0.2 INSERT A

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits.

If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the plant in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.)

The second type of Required Action specifies the remedial measures that permit continued operation of the plant that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "PCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the plant may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

INSERT NEW LCO 3.0.3.BASES

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the plant is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the plant. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the plant in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to enter lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the plant, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Primary Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A plant shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. The LCO is no longer applicable.

c. A Condition exists for which the Required Actions have now been performed.

d. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the

INSERT NEW LCO 3.0.3.BASES

Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in MODE 5 when a shutdown is required during MODE 1 operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for entering the next lower MODE applies. If a lower MODE is entered in less time than allowed, however, the total allowable time to entering MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is entered in 2 hours, then the time allowed for entering MODE 4 is the next 29 hours, because the total time for entering MODE 4 is not reduced from the allowable limit of 31 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to enter a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the plant is already in the most restrictive Condition required by LCO 3.0.3.

The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. Exceptions to LCO 3.0.3 are provided in instances where requiring a plant shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the plant. An example of this is in LCO 3.7.14, "Spent Fuel Pool Water Level."

LCO 3.7.14 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the plant in a shutdown condition. The Required Action of LCO 3.7.14 of "Suspend movement of irradiated fuel assemblies in spent fuel pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

INSERT NEW LCO 3.0.4.BASES

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the plant in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when plant conditions are such that the requirements of the LCO would not be met, in accordance with either LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered following entry into the MODE or other specified condition in the Applicability will permit continued operation within the MODE or other specified condition for an unlimited period of time. Compliance with ACTIONS that permit continued operation of the plant for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the plant before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made and the Required Actions followed after entry into the Applicability.

For example, LCO 3.0.4.a may be used when the Required Action to be entered states that an inoperable instrument channel must be placed in the trip condition within the Completion Time. Transition into a MODE or other specified in condition in the Applicability may be made in accordance with LCO 3.0.4 and the channel is subsequently placed in the tripped condition within the Completion Time, which begins when the Applicability is entered. If the instrument channel cannot be placed in the tripped condition and the subsequent default ACTION ("Required Action and associated Completion Time not met") allows the OPERABLE train to be placed in operation, use of LCO 3.0.4.a is acceptable because the subsequent ACTIONS to be entered following entry into the MODE include ACTIONS (place the OPERABLE train in operation) that permit safe plant operation for an unlimited period of time in the MODE or other specified condition to be entered.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the (continued) Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope.

INSERT NEW LCO
3.0.4.BASES

The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.” Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these system and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a

INSERT NEW LCO 3.0.4.BASES

specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., primary coolant system specific activity), and may be applied to other Specifications based on NRC plant-specific approval.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any plant shutdown. In this context, a plant shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the plant is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

INSERT NEW LCO 3.0.5.BASES

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance. LCO 3.0.5 should not be used in lieu of other practicable alternatives that comply with Required Actions and that do not require changing the MODE or other specified conditions in the Applicability in order to demonstrate equipment is OPERABLE. LCO 3.0.5 is not intended to be used repeatedly.

An example of demonstrating equipment is OPERABLE with the Required Actions not met is opening a manual valve that was closed to comply with Required Actions to isolate a flowpath with excessive Primary Coolant System (PCS) Pressure Isolation Valve (PIV) leakage in order to perform testing to demonstrate that PCS PIV leakage is now within limit.

Examples of demonstrating equipment OPERABILITY include instances in which it is necessary to take an inoperable channel or trip system out of a tripped condition that was directed by a Required Action, if there is no Required Action Note for this purpose. An example of verifying OPERABILITY of equipment removed from service is taking a tripped channel out of the tripped condition to permit the logic to function and indicate the appropriate response during performance of required testing on the inoperable channel. Examples of demonstrating the OPERABILITY of other equipment are taking an inoperable channel or trip system out of the tripped condition;

1) to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system, or

2) to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

The administrative controls in LCO 3.0.5 apply in all cases to systems or components in Chapter 3 of the Technical Specifications, as long as the testing could not be conducted while complying with the Required Actions. This includes the realignment or repositioning of redundant or alternate equipment or trains previously manipulated to comply with

INSERT NEW LCO 3.0.5.BASES

ACTIONS, as well as equipment removed from service or declared inoperable to comply with ACTIONS.

INSERT NEW LCO 3.0.6.BASES

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the plant is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCO's Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.13, "Safety Functions Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained.

INSERT NEW LCO 3.0.6.BASES

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

INSERT NEW LCO
3.0.7.BASES

LCO 3.0.7

Special tests and operations are required at various times over the plant's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, Special Test Exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCO is optional.

A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

Some of the STE LCO require that one or more of the LCO for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCO). The Applicability, ACTIONS, and SRs of the specified normal LCO, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.

Unless the SRs of the specified normal LCO are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCO must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

ACTIONS for STE LCO provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.

INSERT NEW LCO
3.0.8.BASES

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain (continued) capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS). The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for administrative control.

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

Every time that the provisions of LCO 3.0.8 are applied it is required to confirm that at least one train (or subsystem) of systems supported by the inoperable snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. LCO 3.0.8 does not apply to non-seismic snubbers (i.e., seismic vs non-seismic), implementation of this restriction, and the associated plant configuration shall be available on a recoverable basis for NRC staff inspection. SEP-SNB-PLP-001, "Snubber Examination and Testing Program," may be used as a reference for application of LCO 3.0.8 to site specific snubbers.

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

When applying LCO 3.0.8.a at least one AFW train (including a minimum set of supporting equipment required for its successful operation), or some alternative means of core cooling, not associated with the inoperable snubber(s), must be available. Implementation of this restriction and the associated plant configuration shall be available on a recoverable basis for NRC staff inspection.

INSERT NEW LCO 3.0.8.BASES

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

When applying LCO 3.0.8.b at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling (e.g., F&B, fire water system or "aggressive secondary cooldown" using the steam generators) must be available. Implementation of this restriction and the associated plant configuration shall be available on a recoverable basis for NRC staff inspection.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

INSERT NEW LCO
3.0.9.BASES

LCO 3.0.9

LCO 3.0.9 establishes conditions under which systems described in the Technical Specifications are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in the Technical Specifications. This LCO states that the supported system is not considered to be inoperable solely due to required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.9 allows 30 days before declaring the supported system(s) inoperable and the LCO(s) associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that as required barriers are found or are otherwise made unavailable, they are restored. However, the allowable duration may be less than the specified maximum time based on the risk assessment.

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers which are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001).

The provisions of LCO 3.0.9 are justified because of the low risk associated with required barriers not being capable of performing their related support function. This provision is based on consideration of the following initiating event categories:

- Loss of coolant accidents;
- High energy line breaks;
- Feedwater line breaks;
- Internal flooding;
- External flooding;
- Turbine missile ejection; and
- Tornado or high wind.

The risk impact of the barriers which cannot perform their related support function(s) must be addressed pursuant to the risk assessment and management provision of the Maintenance Rule, 10 CFR 50.65 (a)(4), and the associated implementation guidance, Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the

INSERT NEW LCO 3.0.9.BASES

Effectiveness of Maintenance at Nuclear Power Plants." This guidance provides for the consideration of dynamic plant configuration issues, emergent conditions, and other aspects pertinent to plant operation with the barriers unable to perform their related support function(s). These considerations may result in risk management and other compensatory actions being required during the period that barriers are unable to perform their related support function(s).

LCO 3.0.9 may be applied to one or more trains or subsystems of a system supported by barriers that cannot provide their related support function(s), provided that risk is assessed and managed (including consideration of the effects on Large Early Release and from external events). If applied concurrently to more than one train or subsystem of a multiple train or subsystem supported system, the barriers supporting each of these trains or subsystems must provide their related support function(s) for different categories of initiating events. For example, LCO 3.0.9 may be applied for up to 30 days for more than one train of a multiple train supported system if the affected barrier for one train protects against internal flooding and the affected barrier for the other train protects against tornado missiles. In this example, the affected barrier may be the same physical barrier but serve different protection functions for each train.

If during the time that LCO 3.0.9 is being used, the required OPERABLE train or subsystem becomes inoperable, it must be restored to OPERABLE status within 24 hours. Otherwise, the train(s) or subsystem(s) supported by barriers that cannot perform their related support function(s) must be declared inoperable and the associated LCOs declared not met. This 24 hour period provides time to respond to emergent conditions that would otherwise likely lead to entry into LCO 3.0.3 and a rapid plant shutdown, which is not justified given the low probability of an initiating event which would require the barrier(s) not capable of performing their related support function(s). During this 24 hour period, the plant risk associated with the existing conditions is assessed and managed in accordance with 10 CFR 50.65(a)(4).

BASES

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. SR 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 specification.

SR 3.0.1

MODES or other

the OPERABILITY of systems and components, and

SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency.

INSERT SR 3.0.1.A

~~The LCO is assumed to be met when the SRs have been met. Nothing in this Specification, however, is to be construed as implying that the LCO is met when the Surveillance(s) are known to be not met between Surveillance performances.~~

MODE or other

plant

, unless otherwise specified

Surveillances do not have to be performed when the ~~facility~~ is in a specified condition for which the requirements of the associated LCO are not applicable.

INSERT SR 3.0.1.B

INSERT SR 3.0.1.C

~~Surveillances do not have to be performed on variables that are outside their specified limits because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, to restore variables within their specified limits.~~

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances.

plant operating

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers ~~facility~~ conditions that may not be suitable for conducting the Surveillance (e.g., ongoing Surveillance or maintenance activities).

transient conditions or other

and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "Once per . . ." interval

INSERT SR 3.0.1.A

Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

INSERT SR 3.0.1.B

The SRs associated with a Special Test Exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

INSERT SR 3.0.1.C

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary plant parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

An example of this process is:

a. High Pressure Safety Injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2 (continued)

When a Section 5.5, “Programs and Manuals,” specification states that the provisions of SR 3.0.2 are applicable, a 25% extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted.

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs.

INSERT SR 3.0.2.A

(other than those consistent with refueling intervals)

The provisions of SR 3.0.2 are not intended to be used repeatedly to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been performed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

When a Section 5.5, “Programs and Manuals,” specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been completed within the testing interval (including the allowance of SR 3.0.2 if invoked by the Section 5.5 specification).

This delay period provides an adequate time to perform Surveillances that have been missed. This delay period permits the performance of a Surveillance before complying with Required Actions or other remedial measures that might preclude performance of the Surveillance.

INSERT SR 3.0.2.A

The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. Examples of where SR 3.0.2 does not apply are the Containment Leak Rate Testing Program required by 10 CFR 50, Appendix J, and the American Society of Mechanical Engineers (ASME) Code inservice testing required by 10 CFR 50.55a. These programs establish testing requirements and frequencies in accordance with the requirements of regulations. The TS cannot, in and of themselves, extend a test interval specified in the regulations directly or by reference.

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. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

BASES

SR 3.0.3 (continued)

, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.)

The basis for this delay period includes consideration of ~~facility~~ plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified ~~facility~~ unit conditions, is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

SR 3.0.3 is only applicable if there is a reasonable expectation the equipment is OPERABLE or that associated variables are within limits, and it is expected that the Surveillance will be met when performed. Many factors should be considered, such as the period of time since the Surveillance was last performed, or whether the Surveillance, or a portion thereof, has ever been performed, and any other indications, tests, or activities that might support the expectation that the Surveillance will be met when performed.

INSERT SR 3.0.3.A

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used repeatedly to extend Surveillance intervals.

While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on ~~facility~~ plant risk (from delaying the Surveillance as well as any ~~facility~~ plant configuration changes required to perform the Surveillance) and impact on any analysis assumptions, in addition to ~~facility~~ unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. INSERT SR 3.0.3.B All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

INSERT SR 3.0.3.A

An example of the use of SR 3.0.3 would be a relay
contact that was not tested as required in accordance with a particular SR, but previous successful performances of the SR included the relay contact; the adjacent, physically connected relay contacts were tested during the SR performance; the subject relay contact has been tested by another SR; or historical operation of the subject relay contact has been successful. It is not sufficient to infer the behavior of the associated equipment from the performance of similar equipment. The rigor of determining whether there is a reasonable expectation a Surveillance will be met when performed should increase based on the length of time since the last performance of the Surveillance. If the Surveillance has been performed recently, a review of the Surveillance history and equipment performance may be sufficient to support a reasonable expectation that the Surveillance will be met when performed. For Surveillances that have not been performed for a long period or that have never been performed, a rigorous evaluation based on objective evidence should provide a high degree of confidence that the equipment is OPERABLE. The evaluation should be documented in sufficient detail to allow a knowledgeable individual to understand the basis for the determination.

INSERT SR 3.0.3.B

This risk
impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action.

system and component OPERABILITY requirements and

BASES

SR 3.0.3 (continued)

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

MODE or other

systems and components ensure safe operation of the plant

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified Condition in the Applicability.

MODES or other

This Specification ensures that variable limits are met before entry into specified conditions in the Applicability for which these ~~variables ensure safe handling and storage of spent fuel~~.

INSERT SR 3.0.4.A

~~The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring variables within specified limits before entering an associated specified condition in the Applicability.~~

system, subsystem, division, component, device, or

inoperable or

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a variable is outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on ~~variables that are outside their specified limits~~. When ~~a variable is outside its specified limit~~, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.

equipment is inoperable

SR 3.0.4 does not restrict changing specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, providing the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

inoperable equipment

MODES or other

MODES or other

MODE or other

INSERT SR 3.0.4.B

INSERT SR 3.0.4.A

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability. A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to Surveillance not being met in accordance with LCO 3.0.4.

INSERT SR 3.0.4.B

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any plant shutdown. In this context, a plant shutdown is defined as a change in MODE or specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

MODE or other

SR 3.0.4 (continued)

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of maintaining the reactor core subcritical when shut down under cold conditions, in accordance with the Palisades Nuclear Plant design criteria (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown events and Anticipated Operational Occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all full-length control rods, assuming that the single control rod of highest reactivity worth remains fully withdrawn. Once all full-length control rods have been verified to be at or below the lower electrical limit, the penalty for the control rod of highest reactivity worth fully withdrawn no longer must be applied.

The Palisades Nuclear Plant design criteria requires that two separate reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control rods and soluble boric acid in the Primary Coolant System (PCS). The Rod Control System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel design limits, assuming that the control rod of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During MODES 1 and 2, SDM control is ensured by operating with the shutdown rods within the limits of LCO 3.1.5, “Shutdown and Part-Length Rod Group Insertion Limits,” and the regulating rods within the limits of LCO 3.1.6, “Regulating Rod Group Position Limits.” When the plant is in MODES 3, 4, 5, and 6 the SDM requirements are met by means of adjustments to the PCS boron concentration.

BASES

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption that the control rod of highest reactivity worth is fully withdrawn following a reactor trip. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (Departure from Nucleate Boiling Ratio (DNBR), fuel centerline temperature limit AOOs, and ≤ 280 cal/gm energy deposition for the control rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements are based on a Main Steam Line Break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected Steam Generator (SG), and consequently the PCS. This results in a reduction of the primary coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. The most limiting MSLB with respect to potential fuel damage is a guillotine break of a main steam line initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating PCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3 and 4 must also protect against an inadvertent boron dilution; (Ref. 3) and an uncontrolled control rod bank withdrawal from subcritical conditions (Ref. 5).

BASES

APPLICABLE SAFETY ANALYSES (continued)	Each of these events is discussed below. In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the PCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest. The withdrawal of a control rod bank from subcritical conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of control rod banks also produce a time dependent redistribution of core power. Depending on the system initial conditions and reactivity insertion rate, the uncontrolled control rod banks withdrawal transient is terminated by either a high power trip or a high pressurizer pressure trip. In all cases, power level, PCS pressure, linear heat rate, and the DNBR do not exceed allowable limits. SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2).
LCO	The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the value for SDM. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed applicable 10 CFR 50.67 limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable. SDM is a core physics design condition that can be ensured through full-length control rod positioning (regulating and shutdown rods) and through the soluble boron concentration.
APPLICABILITY	In MODE 3, 4 and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, and LCO 3.1.6. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

BASES

ACTIONS

A.1

If the SDM requirements are not met, 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. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the PCS as soon as possible, the boron injection flow should be a highly concentrated solution, such as that normally found in the concentrated boric acid storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the PCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% $\Delta\rho$ must be recovered and a boration flow rate of 35 gpm, it is possible to increase the boron concentration of the PCS by 100 ppm in approximately 25 minutes. If a boron worth of 1.0 E-4 $\Delta\rho$/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta\rho$. These boration parameters of 35 gpm and 100 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SDM is verified by a reactivity balance calculation, considering the listed reactivity effects:

a. PCS boron concentration;

b. Control rod positions;

c. PCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.1.1 (continued)

f. Isothermal Temperature Coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the PCS.

Samarium is not considered in the reactivity analysis since the analysis assumes that the negative reactivity due to Samarium is offset by the positive reactivity of Plutonium built in.

SR 3.1.1.1 requires SDM to be within the limits specified in the COLR. This SDM value ensures the consequences of an MSLB, will be acceptable as a result of a cooldown of the PCS which adds positive reactivity in the presence of a negative moderator temperature coefficient as well as the other events described in the Applicable Safety Analysis. As such, the requirements of this SR must be met whenever the plant is in MODES 3, 4, and 5.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.14
3. FSAR, Section 14.3
4. 10 CFR 50.67
5. FSAR, Section 14.2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

BACKGROUND

According to the Palisades Nuclear Plant design criteria (Ref. 1), reactivity shall be controllable, such that, subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SHUTDOWN MARGIN (SDM) or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons such as burnable absorbers. Excess reactivity can be inferred from the critical boron curve, which provides an indication of the soluble boron concentration in the Primary Coolant System (PCS) versus cycle burnup. Periodic measurement of the PCS boron concentration for comparison with the predicted value with other variables fixed (such as control rod height, temperature, pressure, and power) provides a convenient method of ensuring that core reactivity is within design expectations, and that the calculational models used to generate the safety analysis are adequate.

BASES

BACKGROUND (continued)

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable poisons, full-length control rods, neutron poisons (mainly xenon and samarium) in the fuel, and the PCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the PCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The critical boron curve is based on steady state operation at RTP. Therefore, deviations from the predicted critical boron curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or control rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the PCS boron concentration requirements for reactivity control during fuel depletion.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The comparison between measured and predicted initial core reactivity provides a normalization for calculational models used to predict core reactivity. If the measured and predicted PCS boron concentrations for identical core conditions at Beginning Of Cycle (BOC) are not within design tolerances, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted critical boron curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred. The normalization of predicted PCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle. The reactivity balance satisfies Criterion 2 of 10 CFR 50.36(c)(2).
LCO	The reactivity balance limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta\rho$ has been established, based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

BASES

LCO (continued)

When measured core reactivity is within ± 1% $\Delta\rho$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limits are normally detected by comparing predicted and measured steady state PCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the PCS boron concentration is unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This specification does not apply in MODE 2 because enough operating margin exists to limit the effects of a reactivity anomaly, and THERMAL POWER is low enough (≤ 5% RTP) such that reactivity anomalies are unlikely to occur. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis.

ACTIONS

A.1 and A.2

Should an imbalance develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions.

BASES

ACTIONS

A.1 and A.2 (continued)

The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity imbalance may be resolved. If the cause of the reactivity imbalance is a mismatch in core conditions at the time of PCS boron concentration sampling, then a recalculation of the PCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity imbalance is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the critical boron curve may be renormalized, and power operation may continue. If operational restrictions or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the Required Actions for Condition A are not met within 7 days, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted PCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable including control rod position, moderator temperature, fuel temperature, fuel depletion, and xenon concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note in the Surveillance column which indicates that if the normalization of predicted core reactivity to the measured value is to occur, it must take place within the first 60 Effective Full Power Days (EFPD) after each refueling. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. A second Note, "only required after initial 60 EFPD," is added to the Frequency column to allow this.

REFERENCES

1. FSAR, Section 5.1

2. FSAR, Chapter 14

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to Palisades Nuclear Plant design criteria (Ref. 1), the reactor core and its interaction with the Primary Coolant System (PCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended or rapid reactivity increases.

The MTC relates a change in core reactivity to a change in primary coolant temperature. A positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by the measurement performed as part of startup testing following a refueling. Both initial and reload cores are designed so that the Beginning Of Cycle (BOC) MTC is less positive than that allowed by the LCO. The actual value of the MTC is dependent on core characteristics, such as fuel loading and primary coolant soluble boron concentration. The core design may require additional fixed distributed poisons (lumped burnable poison assemblies) to yield an MTC at BOC within the range analyzed in the plant accident analysis. The End Of Cycle (EOC) MTC is also limited by the requirements of the accident analysis. However, the safety analysis assumptions for the MTC at EOC are assumed by confirming the BOC MTC measurement is within limits which indicates the core is behaving as predicted.

BASES

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and during accidents, such as overheating and overcooling events.

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations, to ensure the accident results are bounding (Ref. 3).

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Examples of reactivity accidents that cause increased power production include the control rod bank withdrawal transient from either partial or RATED THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. Several events discussed in Reference 2 are analyzed with a positive MTC.

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the PCS, and is therefore the most limiting event with respect to the negative MTC, is a Main Steam Line Break (MSLB) event. Following the reactor trip for the postulated EOC MSLB event, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power (approximately 12% RTP) is produced.

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

BASES

LCO

LCO 3.1.3 requires the MTC to be < 0.5 E-4 ⊗ρ/°F at ≤ 2% RTP to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The limit on a positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement. The surveillance check at BOC on the MTC provide confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure startup and subcritical accidents, such as the uncontrolled full-length control rod or group withdrawal, will not violate the assumptions of the accident analysis. The measurement of MTC in MODE 2 prior to exceeding 2% RTP is used to confirm that the core is behaving as analyzed. This ensures that the MTC will remain within the analyzed range while operating in MODES 1 and 2. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of the MTC, with temperature in MODES 3, 4, and 5, for DBAs initiated in MODES 1 and 2, is accounted for in the subject accident analysis. The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC measurement is used for normalization.

BASES

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The SR for measurement of the MTC at the beginning of each fuel cycle provides for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (or least negative) to most negative value during fuel cycle operation as the PCS boron concentration is reduced to compensate for fuel depletion. The requirement for measurement prior to operation > 2% RTP satisfies the confirmatory check on the most positive (or least negative) MTC value. It also confirms that the core is behaving as analyzed which ensures that the MTC will remain within the analysis limits for the remainder of the fuel cycle.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Chapter 14
3. FSAR, Section 3.3

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 allows the operator to detect a control rod that is beginning to deviate from its expected position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 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. 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.) The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.4.4

Demonstrating the rod position deviation alarm is OPERABLE verifies the alarm is functional. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 3.1.5 Shutdown and Part-Length Rod Group Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown rods are initial assumptions in all safety analyses that assume full-length control rod insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected rod worth, and initial reactivity insertion rate.

The Palisades Nuclear Plant design criteria (Ref. 1) and 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors,” contain the applicable criteria for these reactivity and power distribution design requirements. Limits on shutdown rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved.

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

The Palisades Nuclear Plant has four part-length control rods installed. The part-length rods are required to remain completely withdrawn during power operation. The part-length rods do not insert on a reactor trip.

The design calculations are performed with the assumption that the shutdown rod groups are withdrawn prior to the regulating rod groups. The shutdown rods can be fully withdrawn without the core going critical. This provides available negative reactivity for SDM in the event of boration errors. All control rod groups are controlled manually by the control room operator. During normal plant operation, the shutdown rod groups are fully withdrawn. The shutdown rod groups must be completely withdrawn from the core prior to withdrawing any regulating rods during an approach to criticality. The shutdown rod groups are then left in this position until the reactor is shut down.

They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

BASES

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown rod groups are fully withdrawn any time the reactor is critical. This ensures that:

a. The minimum SDM is maintained; and

b. The potential effects of a control rod ejection accident are limited to acceptable limits.

Control rods are considered fully withdrawn at 128 inches, since this position places them in an insignificant reactivity worth region of the integral worth curve for each bank.

On a reactor trip, all full-length control rods (shutdown and regulating), except the most reactive rod, are assumed to insert into the core. The shutdown and regulating rod groups shall be at or above their insertion limits and available to insert the required amount of negative reactivity on a reactor trip signal. The regulating rods may be partially inserted in the core as allowed by LCO 3.1.6, "Regulating Rod Group Position Limits." The shutdown rod group insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)) following a reactor trip from full power. The combination of regulating rod and shutdown rods (less the most reactive rod, which is assumed to remain fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 2). The shutdown rod group insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown rods as well as regulating rod insertion limits and inoperability or misalignment are that:

a. There be no violation of:

   1. Specified acceptable fuel design limits, or

   2. Primary Coolant System pressure boundary damage; and

b. The core remains subcritical after accident transients.

BASES

APPLICABLE SAFETY ANALYSES (continued)

As such, the shutdown and part-length rod group insertion limits affect safety analyses involving core reactivity, ejected rod worth, and SDM (Ref. 2). The part-length control rods have the potential to cause power distribution envelopes to be exceeded if inserted while the reactor is critical. Therefore, they must remain withdrawn in accordance with the limits of the LCO (Ref. 3).

The shutdown and part-length rod group insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The shutdown and part-length rod groups must be within their insertion limits any time the reactor is critical or approaching criticality. For a control rod group to be considered above its insertion limit, all OPERABLE rods in that group, which are not misaligned, must be above the insertion limit (inoperable and misaligned rods are addressed by LCO 3.1.4). Maintaining the shutdown rod groups within their insertion limits ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. Maintaining the part-length rod group within its insertion limit ensures that the power distribution envelope is maintained.

APPLICABILITY

The shutdown and part-length rod groups must be within their insertion limits, with the reactor in MODES 1 and 2. In MODE 2 the Applicability begins anytime any regulating rod is withdrawn above 5 inches. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODE 4, 5, or 6, the shutdown rod groups are inserted in the core to at least the lower electrical limit and contribute to the SDM. In MODE 3 the shutdown rod groups may be withdrawn in preparation of a reactor startup. Refer to LCO 3.1.1, “SHUTDOWN MARGIN (SDM),” for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.3 (rod exercise test). Control rod exercising verifies the freedom of the rods to move, and requires the individual shutdown rods to move below the LCO limits for their group. Only the full-length rods are required to be tested by SR 3.1.4.3. The part-length rods may also be moved however, if a part-length rod is moved below the limit of the associated LCO, the Required Actions of Condition A must be taken. Positioning of an individual control rod within its group is addressed by LCO 3.1.4, “Control Rod Alignment.”

BASES

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.

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 $\leq$ 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).

BASES

ACTIONS (continued)

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.

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.

BASES

ACTIONS (continued)

A.1

Prior to entering this condition, the shutdown and part-length rod groups were fully withdrawn. If a shutdown rod group is then inserted into the core, its potential negative reactivity is added to the core as it is inserted.

If one or more shutdown or part-length rods are not within limits, the affected rod(s) must be declared inoperable and the applicable Conditions and Required Actions of LCO 3.1.4 entered immediately. This Required Action is based on the recognition that the shutdown and part-length rods are normally withdrawn beyond their insertion limits and are capable of being moved by their control rod drive mechanism. Although the requirements of this LCO are not applicable during performance of the control rod exercise test, the inability to restore a control rod to within the limits of the LCO following rod exercising would be indicative of a problem affecting the OPERABILITY of the control rod. Therefore, entering the applicable Conditions and Required Actions of LCO 3.1.4 is appropriate since they provide the applicable compensatory measures commensurate with the inoperability of the control rod.

B.1

When Required Action A.1 cannot be met or 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 3.1.5.1

Verification that the shutdown and part-length rod groups are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown rods will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. Verification that the part-length rod groups are within their insertion limits ensures that they do not adversely affect power distribution requirements. This SR ensures that the shutdown and part-length rod groups are withdrawn before the regulating rods are withdrawn during a plant startup.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.2
3. FSAR, Section 14.6

INSERT Bases 3.1.6

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Regulating Rod Group Position Limits

BASES

BACKGROUND

The insertion limits of the regulating rod groups are initial assumptions in all safety analyses that assume full-length rod insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are contained in the Palisades Nuclear Plant design criteria (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating rod group insertion have been established, and all regulating rod group positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected rod worth, reactivity insertion rate, and SDM limits are preserved.

The regulating rod groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between rod worth and rod position (integral rod worth). The regulating rod groups are withdrawn and operate in a predetermined sequence. The group sequence and overlap limits are specified in the COLR.

The regulating rods are used for precise reactivity control of the reactor. The positions of the regulating rods are manually controlled. They are capable of changing reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.6; LCO 3.2.3, "QUADRANT POWER TILT (T_q)"; and LCO 3.2.4, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LCO 3.2.1, "Linear Heat Rate (LHR)") and F_R^T (LCO 3.2.2, "TOTAL RADIAL PEAKING FACTOR (F_R^T)") limits in the COLR.

BASES

BACKGROUND (continued)

Operation within the LHR limits given in the COLR prevents power peaks that would exceed the Loss Of Coolant Accident (LOCA) limits derived by the Emergency Core Cooling System analysis. Operation within the F_R^T limits given in the COLR prevents Departure from Nucleate Boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the LHR and F_R^T limits, certain reactivity limits are preserved by regulating rod insertion limits. The regulating rod group insertion limits also restrict the ejected rod worth to the values assumed in the safety analysis and preserve the minimum required SDM in MODES 1 and 2.

The ejected rod case is limited to the reactivity worth for the highest worth rod ejected from the PDIL limit, thus limiting the maximum possible reactivity excursion.

The establishment of limiting safety system settings and LCOs requires that the expected long and short term behavior of the F_R^T be determined. The long term behavior relates to the variation of the steady state F_R^T with core burnup and is affected by the amount of rod insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks, due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the rods during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). The PDIL curve stated in the COLR dictates the acceptable regulating rod group positioning for anticipated power maneuvers and transient mitigation within the limits. The PDIL limitations stated in the COLR reflect the assumptions made in the safety analyses. This ensures that the F_R^T limits are not violated during power level maneuvering or transient mitigation.

The regulating rod group insertion and alignment limits are process variables that together characterize and control the three-dimensional power distribution of the reactor core. Additionally, the regulating rod group insertion limits control the reactivity that could be added in the event of a control rod ejection accident, and the shutdown and regulating bank insertion limits ensure the required SDM is maintained.

BASES

BACKGROUND (continued)

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The acceptance criteria for the regulating rod group position, ASI, and T_q LCOs are such as to preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F, (Ref. 2);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. The rods must be capable of shutting down the reactor with a minimum required SDM, with the highest worth rod stuck fully withdrawn (Ref. 1).

Regulating rod group position, ASI, and T_q are process variables that together characterize and control the three-dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SDM requirement is ensured by limiting the regulating and shutdown rod group insertion limits, so that the allowable inserted worth of the rods is such that sufficient reactivity is available to shut down the reactor to hot zero power. SDM assumes the maximum worth rod remains fully withdrawn upon trip (Ref. 4).

The most limiting SDM requirements for Mode 1 and 2 conditions at Beginning of Cycle (BOC) are determined by the requirements of several transients, e.g., Loss of Flow, etc. However, the most limiting SDM requirements for MODES 1 and 2 at End of Cycle (EOC) come from just one transient, Main Steam Line Break (MSLB). The requirements of the MSLB event at EOC for the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

Although the most limiting SDM requirements at EOC are much larger than those at BOC, the available SDMs obtained via tripping the full-length control rods are substantially larger due to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle are sufficient since the difference between available SDMs and the limiting SDM requirements are the smallest at these times in cycle. The measurement of full-length control rod bank worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adherence to LCO 3.1.5, “Shutdown and Part-Length Rod Group Insertion Limits,” and LCO 3.1.6 provides assurance that the available SDM at any time in cycle will exceed the limiting SDM requirements at that time in cycle.

Operation at the insertion limits or ASI limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed Tq present. Operation at the insertion limit may also indicate the maximum ejected rod worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected rod worth.

The regulating and shutdown rod insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected rod worth, and peaking factors are preserved.

The regulating rod group position limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

BASES

LCO

The limits on regulating rod group sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating rod groups provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod group motion. For a control rod group to be considered above its insertion limit, all OPERABLE rods in that group, which are not misaligned, must be above the insertion limit (inoperable and misaligned rods are addressed by LCO 3.1.4).

The Power Dependent Insertion Limit (PDIL) alarm circuit is required to be OPERABLE for notification that the regulating rod groups are outside the required insertion limits. The Control Rod Out Of Sequence (CROOS) alarm circuit is required to be OPERABLE for notification that the rods are not within the required sequence and overlap limits. When the PDIL or the CROOS alarm circuit is inoperable, the verification of rod group positions is increased to ensure improper rod alignment is identified before unacceptable flux distribution occurs. The PDIL and CROOS alarms can be generated by either the synchro based Primary Indication Processor (PIP) node, or the reed switch based Secondary Position Indication (SPI) system since the SPI system, in conjunction with the host computer, is redundant to the PIP node in the task of control rod measurement, control rod monitoring and limit processing.

APPLICABILITY

The regulating rod group sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES. SDM is preserved in MODES 3, 4, and 5 by adjustments to the soluble boron concentration.

The Applicability has been modified by a Note indicating the LCO requirement is suspended while performing SR 3.1.4.3 (rod exercise test). Control rod exercising verifies the freedom of the rods to move, and requires the individual regulating rods to move below the LCO limits which could violate the LCO for their group.

BASES

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.

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).

BASES

ACTIONS (continued)

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.

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.

BASES

ACTIONS (continued)

A.1 and A.2

Operation beyond the insertion limit may result in a loss of SDM and excessive peaking factors. The insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the regulating rods in response to changing plant conditions.

When the regulating groups are inserted beyond the insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual rod group position limit. Two hours provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

B.1

Operating outside the regulating rod group sequence and overlap limits specified in the COLR may result in excessive peaking factors. If the sequence and overlap limits are exceeded, the regulating rod groups must be restored to within the appropriate sequence and overlap. Two hours provides adequate time for the operator to restore the regulating rod group to within the appropriate sequence and overlap limits.

C.1

When the PDIL or the CROOS alarm circuit is inoperable, performing SR 3.1.6.1 once within 15 minutes following any rod motion ensures improper rod alignments are identified before unacceptable flux distributions occur.

D.1

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 3.1.6.1

With the PDIL alarm circuit OPERABLE, verification of each regulating rod group position is sufficient to detect rod positions that may approach the acceptable limits, and to provide the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.6.2

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.6.3

Demonstrating the CROOS alarm circuit OPERABLE verifies that the CROOS alarm circuit is functional. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 5.1
2. 10 CFR 50.46
3. FSAR, Section 14.16
4. FSAR, Section 14.4

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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;

e. Xenon concentration; and

f. Isothermal Temperature Coefficient (ITC).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.3

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Linear Heat Rate (LHR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the primary coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected control rod accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using control rods to alter the axial power distribution;

b. Decreasing control rod insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a control rod drop or misoperation of the plant) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., control rod insertion and alignment limits), the power distribution satisfies this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution changes over time also minimizes the xenon distribution changes, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions.

BASES

BACKGROUND (continued)

The limits on LHR, TOTAL RADIAL PEAKING FACTOR (F_R^T), QUADRANT POWER TILT (T_q), and AXIAL SHAPE INDEX (ASI), which are obtained directly from the core reload analysis, ensure compliance with the safety limits on LHR and Departure from Nucleate Boiling Ratio (DNBR).

Either of the two core power distribution monitoring systems, the Incore Alarm portion of the Incore Monitoring System or the Excore Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR is within its limits. The Incore Alarm System performs this function by continuously monitoring the local power at many points throughout the core and comparing the measurements to predetermined setpoints above which the limit on LHR could be exceeded. The Excore Monitoring System performs this function by providing comparison of the measured core ASI with predetermined ASI limits based on incore measurements. An Excore Monitoring System Allowable Power Level (APL), which may be less than RATED THERMAL POWER, and an additional restriction on T_q, are applied when using the Excore Monitoring System to ensure that the ASI limits adequately restrict the LHR to less than the limiting values.

In conjunction with the use of the Excore Monitoring System for monitoring LHR and in establishing ASI limits, the following assumptions are made:

a. The control rod 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," are satisfied;

b. The additional T_q restriction of SR 3.2.1.6 is satisfied; and

c. F_R^T, does not exceed the limits of LCO 3.2.2.

The limitations on the TOTAL RADIAL PEAKING FACTOR provided in the COLR ensure that the assumptions used in the analysis for establishing the LHR limits and Limiting Safety System Settings (LSSS) remain valid during operation at the various allowable control rod group insertion limits.

BASES

BACKGROUND (continued)

The Incore Monitoring System continuously provides a direct indication of the core power distribution. It also provides alarms that have been established for the individual incore detector segments, ensuring that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, for:

a. A measurement calculational uncertainty factor (as identified in the COLR);

b. An engineering uncertainty factor of 1.03; and

c. A THERMAL POWER measurement uncertainty factor of 1.006 of 2565.4 MWt.

The measurement uncertainties associated with LHR and F_R^T are based on a statistical analysis performed on power distribution benchmarking results. The COLR includes the applicable measurement uncertainties for incore detector usage. The engineering and THERMAL POWER uncertainties are incorporated in the power distribution calculation performed by the fuel vendor.

The excore power distribution monitoring system consists of Power Range Channels 5 through 8. The power range channels monitor neutron flux from 0 to 125 percent full power. They are arranged symmetrically around the reactor core to provide information on the radial and axial flux distributions.

The power range detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, which is located directly above it, monitors flux from the upper half of the core. The DC current signal from each of the ion chambers is fed directly to the control room drawer assembly without pre-amplification. Each excore detector supplies data to a Thermal Margin Monitor (TMM). Each TMM uses these excore signals to calculate Axial Shape Index (ASI) on a continuous basis.

ASI can be defined as the compensated ratio of power developed in the upper and lower sections of the core. The TMM takes the excore detector signals and develops a power ratio (YE) that describes the distribution of neutron flux developed in the core by the formula:

$$\mathbf{YE = (L - U)/(L + U)}$$

Where L is the lower excore segment flux, and U is the upper excore segment flux.

BASES

BACKGROUND (continued)

The excore detectors which are located within the concrete biological shield of the reactor must be compensated for the phenomenon of shape annealing. Shape annealing factors are developed to correct the excore readings for neutron attenuation from the core periphery to the excore detector locations. This accounts for any material that would cause neutron attenuation within the detector path such as: concrete, structural steel and so forth. This allows the excore detectors to represent an accurate measurement of the core power distribution. Shape annealing has been found to be a linear relationship which can be correlated to the Axial Offset (AO) as determined by an Incore Detector System to the raw readings seen by the excore detectors.

Reactor Engineering has developed shape annealing factors for each individual Excore detector. The TMM uses the above calculated power ratio and the appropriate shape annealing factor to determine the ASI value for an individual excore detector channel.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or AOOs (Condition 2) (Ref. 3). The power distribution and control rod insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 4);

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 3).

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm; and

d. The full-length control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This is accomplished by maintaining the power distribution and primary coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by accident analyses (Ref. 1), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 4). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Primary Coolant System Operation ensure that these criteria are met as long as the core is operated within the LHR, ASI, F_R^T, and T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not necessarily occur while the plant is operating at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident occurs from initial conditions outside the limits of these LCOs. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR.

The Incore Monitoring System provides for monitoring of LHR, F_R^T, and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained. The Incore Monitoring System is also utilized to determine the target AXIAL OFFSET (AO) and to determine the Allowable Power Level (APL) when using the excore detectors.

The Excore Monitoring System provides for monitoring of ASI and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained.

LHR satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on the LHR in the peak power fuel rod at the peak power elevation Z ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

BASES

LCO (continued)	The LCO requires that LHR be maintained within the limits specified in the COLR and either the Incore Alarm System or Excore Monitoring System be OPERABLE to monitor LHR. When using the Incore Alarm System, the LHR is not considered to be out of limits until there are four or more incore detectors simultaneously in alarm. When using the Excore Monitoring System, LHR is considered within limits when the conditions are acceptable for use of the Excore Monitoring System and the associated ASI and T_q limits specified in the SRs are met. To be considered OPERABLE, the Incore Alarm System must have at least 90 of the 180 incore detectors OPERABLE and 2 incore detectors per axial level per core quadrant OPERABLE. In addition, the plant process computer must be OPERABLE and the required alarm setpoints entered into the plant computer. Only 36 of the 45 instrument locations are included in the Incore Alarm System Uncertainty Analysis (180 of the possible 215 detectors). Instrument locations 1, 4, 13, 34, 41, 42 and 45 are not included, and instrument locations 7 and 44 are used by the Reactor Vessel Level Monitoring System (RVLMS). To be considered OPERABLE, the Excore Monitoring System must have been calibrated with OPERABLE incore detectors, the ASI must not have been out of limits for the last 24 hours, and THERMAL POWER must be less than the APL.
APPLICABILITY	In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER ≤ 25% RTP, and in other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution, and because ample thermal margin exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions remain valid.
ACTIONS	A.1 There are three acceptable methods for verifying that LHR is within limits. The LCO requires monitoring by either an OPERABLE Incore Alarm System or an OPERABLE Excore Monitoring System. When both of the required systems are inoperable, Condition B allows for monitoring by taking manual readings of the incore detectors. Any of these three methods may indicate that the LHR is not within limits. With the LHR exceeding its limit, excessive fuel damage could occur following an accident. In this Condition, prompt action must be taken to restore the LHR to within the specified limits. One hour to restore the LHR to within its specified limits is reasonable and ensures that the core does not continue to operate in this Condition. The 1 hour

BASES

ACTIONS (Continued)

A.1

Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

ACTIONS (continued)

B.1 and B.2

With the Incore Alarm System inoperable for monitoring LHR and the Excore Monitoring System inoperable for monitoring LHR, THERMAL POWER must be reduced to ≤ 85% RTP within 2 hours. Operation at ≤ 85% RTP ensures that ample thermal margin is maintained. A 2 hour Completion Time is adequate to achieve the required plant condition without challenging plant systems. Additionally, with the Incore Alarm and Excore Monitoring Systems inoperable, LHR must be verified to be within limits within 4 hours, and every 2 hours thereafter by manually collecting incore detector readings at the terminal blocks in the control room utilizing a suitable signal detector. The manual readings shall be taken on a minimum of 10 individual detectors per quadrant (to include a total of 90 detectors in a 10 hour period). The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the power distribution to detect significant changes until the monitoring systems are returned to service.

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 3.2.1 B.2 must be initially performed within 4 hours without any SR 3.0.2 extension, subsequent performances at the “Once per 2 hours” interval may utilize the 25% SR 3.0.2 extension.

C.1

If the Required Action and associated Completion Time are not met, THERMAL POWER must be reduced to ≤ 25% RTP. This reduced power level ensures that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reach ≤ 25% RPT from full power MODE 1 conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

The Incore Alarm portion of the Incore Monitoring System provides continuous monitoring of LHR through the plant computer. The PIDAL computer program is used to generate alarm setpoints for the plant computer that are based on measured margin to allowed LHR. As the incore detectors are read by the plant computer, they are continuously compared to the alarm setpoints. If the Incore Alarm System LHR monitoring function is inoperable, excore detectors or manual recordings of the incore detector readings may be used to monitor LHR. Periodically monitoring LHR ensures that the assumptions made in the Safety Analysis are maintained. This SR is modified by a Note that states that the SR is only required to be met when the Incore Alarm System is being used to monitor LHR. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.2.1.2

Continuous monitoring of the LHR is provided by the Incore Alarm System which provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its specified limits.

Performance of this SR verifies the Incore Alarm System can accurately monitor LHR by ensuring the alarm setpoints are based on a measured power distribution. Therefore, they are only applicable when the Incore Alarm System is being used to determine the LHR.

The alarm setpoints must be initially adjusted following each fuel loading prior to operation above 50% RTP, and periodically adjusted thereafter. The SR is modified by a Note which requires the SR to be met only when the Incore Alarm System is being used to determine LHR. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.3

SR 3.2.1.3 requires, prior to initial use of the excore LHR monitoring function, verification that the absolute difference of the measured ASI and the target ASI has been ≤ 0.05 for each OPERABLE channel for the last 24 hours using the previous 24 hourly recorded values. Performance of this SR verifies that plant conditions are acceptable for the Excore Monitoring System to accurately monitor the LHR (Ref. 5). The prior to initial use verification identifies that there have been no significant power distribution anomalies while using other monitoring methods, e.g., the incore detectors, which may affected the ability of the excore detectors to monitor LHR.

The SR is modified by a Note that states that the SR is only required to be met when the Excore Monitoring System is being used to monitor LHR. Failure of this SR prevents the Excore Monitoring System from being considered OPERABLE for monitoring of LHR.

SR 3.2.1.4

SR 3.2.1.4 requires verification that THERMAL POWER is less than or equal to the Allowable Power Level (APL) which is limited to not more than 10% greater than the THERMAL POWER at which the APL was last determined. Performance of this SR also verifies that plant conditions are acceptable for the Excore Monitoring System to accurately monitor the LHR (Ref. 5). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that states that the SR is only required to be met when the Excore Monitoring System is being used to monitor LHR. Failure of this SR prevents the Excore Monitoring System from being considered OPERABLE for monitoring of LHR.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.5

SR 3.2.1.5 requires verification that the absolute difference of the measured ASI and the target ASI is ≤ 0.05 every hour. This must be verified on at least 3 of the 4, 2 of the 3, or 2 of the 2 OPERABLE channels, whichever is the applicable case. However, any otherwise OPERABLE channel which indicates an absolute difference of > 0.05 must be considered out of limits. Performance of this SR verifies that plant conditions are acceptable for the Excore Monitoring System to be used to assure LHR is within limits (Ref. 5). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that states that the SR is only required to be met when the Excore Monitoring System is being used to monitor LHR. Failure of this SR (when using an OPERABLE Excore Monitoring System) is a failure to verify that LHR is within limits and is therefore considered a failure to meet the LCO due to LHR not within limits as determined by the Excore Monitoring System.

SR 3.2.1.6

SR 3.2.1.6 requires verification that the QUADRANT POWER TILT is ≤ 0.03. Performance of this SR also verifies that plant conditions are acceptable for the Excore Monitoring System to be used to assure LHR is within limits (Ref. 5). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that states that the SR is only required to be met when the Excore Monitoring System is being used to monitor LHR. Failure of this SR (when using an OPERABLE Excore Monitoring System) is a failure to verify that LHR is within limits and is therefore considered a failure to meet the LCO due to LHR not within limits as determined by the Excore Monitoring System.

BASES

REFERENCES

1. FSAR, Chapter 14

2. FSAR, Chapter 6

3. FSAR, Section 5.1

4. 10 CFR 50.46

5. Safety Evaluation Report for Palisades Nuclear Plant Operating License Amendment No. 68, Section 4, dated December 8, 1981

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 TOTAL RADIAL PEAKING FACTOR (F_R^T)

BASES

BACKGROUND	The Background section of Bases B 3.2.1, "Linear Hear Rate," is applicable to these Bases.
APPLICABLE SAFETY ANALYSES	The Applicable Safety Analyses section of Bases B 3.2.1 is applicable to these Bases. The TOTAL RADIAL PEAKING FACTOR satisfies Criterion 2 of 10 CFR 50.36(c)(2).
LCO	The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits, except Tq, are provided in the COLR. The limitations on F_R^T are provided to ensure that assumptions used in the analysis for establishing DNB margin, LHR limit and the thermal margin/low pressure and variable high power trip setpoints remain valid during operation. Data from the incore detectors are used for determining the measured F_R^T.
APPLICABILITY	In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER ≤ 25% RTP, and in other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution, and because ample thermal margin exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions remain valid.

BASES

ACTIONS

A.1

If F_R^T exceeds its limits, F_R^T must be restored to within the limits as identified in the COLR within 6 hours. Restoration may be either by correcting the source of the peaking or by a reduction in THERMAL POWER. The THERMAL POWER typically necessary to achieve restoration is identified by the equation:

$$\mathbf{P = [1\text{-}3.33\ ((F_R/F_L)\text{-}1)]\ (RTP)}$$

Where FR is the measured value of F_R^T; and F_L is the corresponding limit provided in the COLR. Operating at or below this power level, P, is typically sufficient to restore F_R^T within limits. If power reductions do not restore F_R^T to within limits within 6 hours, Condition B is applicable.

Six hours to restore F_R^T to within limit(s) is reasonable and ensures that the core does not continue to operate in this condition for an extended period. The 6 hour Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

B.1

If the Required Action and associated Completion Time are not met, THERMAL POWER must be reduced to ≤ 25% RTP. This reduced power level ensures that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reach ≤ 25% RTP from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The periodic Surveillance to determine F_R^T ensures that F_R^T remains within the range assumed in the analysis throughout the fuel cycle. Determining F_R^T using the incore detectors after each fuel loading prior to the reactor exceeding 50% RTP ensures that the core is properly loaded.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

None

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 QUADRANT POWER TILT (T_q)

BASES

BACKGROUND

The Background section for Bases B 3.2.1, “Linear Heat Rate,” is applicable to these Bases, with the following addition:

The power range monitoring system provides alarms when T_q exceeds predetermined values. The average of the four power range signals is developed by a single “Comparitor Averager.” Each power range channel compares its output signal to this average signal. Two channel deviation alarm bistables, set at different levels, are provided in each power range channel. The deviation alarms will annunciate when the associated channel signal is either above or below the average, however, only a signal above the average is of concern with regard to T_q.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases B 3.2.1 is applicable to these Bases.

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

LCO

The power distribution LCO limits are based on correlations between power peaking and the measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limits on T_q ensure that assumptions used in the analysis for establishing LHR limits and DNB margin remain valid during operation.

APPLICABILITY

In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in accident analysis to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER ≤ 25% RTP, and in other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on core power distribution, and because ample thermal margin exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions remain valid.

BASES

ACTIONS

A.1

If the measured T_q is > 0.05, Tq must be restored within 2 hours or F_R^T must be determined to be within the limits of LCO 3.2.2, and determined to be within these limits every 8 hours thereafter, as long as T_q is out of limits. Two hours is sufficient time to allow the operator to reposition control rods, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in F_R^T can be identified before the limits of LCO 3.2.2 are exceeded.

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 3.2.3 A.1 must be initially performed within 2 hours without any SR 3.0.2 extension, subsequent performances at the "Once per 8 hours" interval may utilize the 25% SR 3.0.2 extension.

B.1

With the measured T_q > 0.10, power must be reduced to < 50% RTP within 4 hours, FRT must be within specified limits to ensure that acceptable flux peaking factors are maintained as required by Condition A (which continues to be applicable). Based on operating experience, 4 hours is sufficient time for evaluation of these factors. If F_R^T is within limits, operation may proceed while attempts are made to restore T_q to within its limit. If the tilt is generated due to a control rod misalignment, continued operation at < 50% RTP allows for realignment; if the cause is other than control rod misalignment, continued operation may be necessary to discover the cause of the tilt. Reducing THERMAL POWER to < 50% RTP, and the more frequent measurement of F_R^T required by ACTION A.1, provide conservative protection from potential increased peaking due to xenon redistribution.

BASES

ACTIONS (continued)

C.1

If T_q is > 0.15, or if Required Actions and associated Completion Times are not met, THERMAL POWER must be reduced to ≤ 25% RTP. This requirement ensures that the core is operating within its thermal limits and places the core in a conservative condition. Four hours is a reasonable time to reach 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

QUADRANT POWER TILT (T_q) is determined from excore detector readings which are calibrated using incore detector measurements (Ref. 1). Calibration factors are determined using incore measurements and an incore analysis computer program (Ref. 2). Each power range channel provides alarms if T_q exceeds its limits. Therefore, with all power range channels OPERABLE, this SR only requires verification that the channel deviation alarms do not indicate an excessive T_q. If the Excore Monitoring System T_q deviation alarm monitoring function is inoperable, excore detector readings or symmetric incore detector readings may be used to monitor T_q at 12 hour intervals. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 7.6.2.2

2. FSAR, Section 7.6.2.4

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 AXIAL SHAPE INDEX (ASI)

BASES

BACKGROUND

The Background section for Bases B 3.2.1, “Linear Heat Rate,” is applicable to these Bases, with the following addition:

The Excore Monitoring System ASI alarm function consists of four channels. At least two channels of the ASI alarm function are necessary to verify that ASI is within limits. With one or more excore monitoring channels measured ASI differing from the incore measured AO by > 0.02 under steady state operating conditions, the ASI monitoring channel alarm setpoint may be adjusted to compensate for this deviation. This ensures that fuel design parameters can continue to be accurately monitored and not exceeded when the incore/excore alignment is not within normal tolerances. This may occur when the calibration cannot be performed or the alignment problem exists after the calibration.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section for Bases B 3.2.1 is applicable to these Bases.

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

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. These power distribution LCO limits, except T_q, are provided in the COLR. The limitation on ASI ensures that the assumed axial power profiles used in the development of the inlet temperature LCO bound the measured axial power profile.

The limitation on ASI, along with the limitations of LCO 3.3.1, "Reactor Protective System Instrumentation," represents a conservative envelope of operating conditions consistent with the assumptions that have been analytically demonstrated adequate for maintaining an acceptable minimum DNBR throughout all AOOs. Operation of the core with conditions within the specified limits ensures that an acceptable minimum margin from DNB conditions is maintained in the event of any AOO.

BASES

APPLICABILITY

In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER ≤ 25% RTP, and in other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution, and because ample thermal margin exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions remain valid.

ACTIONS

A.1

Operating the core within ASI limits specified in the COLR and within the limits of LCO 3.3.1 ensures an acceptable margin for DNB and for maintaining local power density in the event of an AOO. Maintaining ASI within limits also ensures that the limits of 10 CFR 50.46 are not exceeded during accidents. The Required Actions to restore ASI must be completed within 2 hours to limit the duration the plant is operated outside the initial conditions assumed in the accident analyses. In addition, this Completion Time is sufficiently short that the xenon distribution in the core cannot change significantly.

B.1

If the Required Action and associated Completion Time are not met, core power must be reduced. Reducing THERMAL POWER to ≤ 25% RTP ensures that the core is operating farther from thermal limits and places the core in a conservative condition. Four hours is a reasonable amount of time, based on operating experience, to reduce THERMAL POWER to ≤ 25% RTP in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

Verifying that the ASI is within the limits specified in the COLR ensures that the core is not approaching DNB conditions. ASI is determined from excore detector readings which are calibrated using incore detector measurements (Ref. 1). Calibration factors are determined using incore measurements and an incore analysis computer program (Ref. 2). ASI is normally calculated and compared to the alarm setpoints continuously and automatically. Therefore, this SR only requires verification that alarms do not indicate an excessive ASI. If the Excore Monitoring System ASI Alarm function is inoperable, excore detector or incore indications may be used to monitor ASI. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 7.6.2.2

2. FSAR, Section 7.6.2.4

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protective System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the acceptable fuel design limits and breaching the reactor coolant pressure boundary during Anticipated Operational Occurrences (AOOs). (As defined in 10 CFR 50, Appendix A, "Anticipated operational occurrences 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.") By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Values, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The Primary Coolant System (PCS) pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within applicable 10 CFR 50.67 (Ref. 6) and 10 CFR 100 (Ref. 2) criteria during AOOs.

BASES

BACKGROUND (continued)

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within applicable 10 CFR 50.67 (Ref. 6) and 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- RPS trip units;
- Matrix Logic; and
- Trip Initiation Logic.

This LCO addresses measurement channels and RPS trip units. It also addresses the automatic bypass removal feature for those trips with Zero Power Mode bypasses. The RPS Logic and Trip Initiation Logic are addressed in LCO 3.3.2, "Reactor Protective System (RPS) Logic and Trip Initiation." The role of the measurement channels, RPS trip units, and RPS Bypasses is discussed below.

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.

With the exception of High Startup Rate, which employs two instrument channels, and Loss of Load, which employs a single pressure sensor, four identical measurement channels with electrical and physical separation are provided for each parameter used in the direct generation of trip signals. These are designated channels A through D. Some measurement channels provide input to more than one RPS trip unit within the same RPS channel. In addition, some measurement channels may also be used as inputs to Engineered Safety Features (ESF) bistables, and most provide indication in the control room.

BACKGROUND (continued)

Measurement Channels (continued)

In the case of High Startup Rate and Loss of Load, where fewer than four sensor channels are employed, the reactor trips provided are not relied upon by the plant safety analyses. The sensor channels do however, provide trip input signals to all four RPS channels.

When a channel monitoring a parameter exceeds a predetermined setpoint, indicating an abnormal condition, the bistable monitoring the parameter in that channel will trip. Tripping two or more channels of bistable trip units monitoring the same parameter de-energizes Matrix Logic, (addressed by LCO 3.3.2) which in turn de-energizes the Trip Initiation Logic. This causes all four DC clutch power supplies to de-energize, interrupting power to the control rod drive mechanism clutches, allowing the full length control rods to insert into the core.

For those trips relied upon in the safety analyses, three of the four measurement and trip unit channels can meet the redundancy and testability of GDC 21 in 10 CFR 50, Appendix A (Ref. 1). This LCO requires, however, that four channels be OPERABLE. The fourth channel provides additional flexibility by allowing one channel to be removed from service (trip channel bypassed) for maintenance or testing while still maintaining a minimum two-out-of-three logic.

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

Most of the RPS trips are generated by comparing a single measurement to a fixed bistable setpoint. Two trip Functions, Variable High Power Trip and Thermal Margin Low Pressure Trip, make use of more than one measurement to provide a trip.

The required RPS Trip Functions utilize the following input instrumentation:

- Variable High Power Trip (VHPT)

  The VHPT uses Q Power as its input. Q Power is the higher of NI power from the power range NI drawer and primary calorimetric power (ΔT power) based on PCS hot leg and cold leg temperatures. The measurement channels associated with the VHPT are the power range excore channels, and the PCS hot and cold leg temperature channels.

BASES

BACKGROUND (continued)

Measurement Channels

- Variable High Power Trip (VHPT) (continued)

  The Thermal Margin Monitors provide the complex signal processing necessary to calculate the TM/LP trip setpoint, VHPT trip setpoint and trip comparison, and Q Power calculation. On power decreases the VHPT setpoint tracks power levels downward so that it is always within a fixed increment above current power, subject to a minimum value.

  On power increases, the trip setpoint remains fixed unless manually reset, at which point it increases to the new setpoint, a fixed increment above Q Power at the time of reset, subject to a maximum value. Thus, during power escalation, the trip setpoint must be repeatedly reset to avoid a reactor trip.

- High Startup Rate Trip

  The High Startup Rate trip uses the wide range Nuclear Instruments (NIs) to provide an input signal. There are only two wide range NI channels. The wide range channel signal processing electronics are physically mounted in RPS cabinet channels C (NI-1/3) and D (NI-2/4). Separate bistable trip units mounted within the NI-1/3 wide range channel drawer supply High Startup Rate trip signals to RPS channels A and C. Separate bistable trip units mounted within the NI-2/4 wide range channel drawer provide High Startup Rate trip signals to RPS channels B and D.

- Low Primary Coolant Flow Trip

  The Low Primary Coolant Flow Trip utilizes 16 flow measurement channels which monitor the differential pressure across the primary side of the steam generators. Each RPS channel, A, B, C, and D, receives a signal which is the sum of four differential pressure signals. This totalized signal is compared with a setpoint in the RPS Low Flow bistable trip unit for that RPS channel.

BASES

BACKGROUND (continued)

Measurement Channels (continued)

- Low Steam Generator Level Trips

  There are two separate Low Steam Generator Level trips, one for each steam generator. Each Low Steam Generator Level trip monitors four level measurement channels for the associated steam generator, one for each RPS channel.

- Low Steam Generator Pressure Trips

  There are also two separate Low Steam Generator Pressure trips, one for each steam generator. Each Low Steam Generator Pressure trip monitors four pressure measurement channels for the associated steam generator, one for each RPS channel.

- High Pressurizer Pressure Trip

  The High Pressurizer Pressure Trip monitors four pressurizer pressure channels, one for each RPS channel.

- Thermal Margin Low Pressure (TM/LP) Trip

  The TM/LP Trip utilizes bistable trip units. Each of these bistable trip units receives a calculated trip setpoint from the Thermal Margin Monitor (TMM) and compares it to the measured pressurizer pressure signal. The TM/LP setpoint is based on Q power (the higher of NI power from the power range NI drawer, or ΔT power, based on PCS hot leg and cold leg temperatures) pressurizer pressure, PCS cold leg temperature, and Axial Shape Index. The TMM provide the complex signal processing necessary to calculate the TM/LP trip setpoint, TM/LP trip comparison signal, and Q Power.

BASES

BACKGROUND (continued)

Measurement Channels (continued)

- Loss of Load Trip

  The Loss of Load Trip is initiated by two-out-of-three logic from pressure switches in the turbine auto stop oil circuit that sense a turbine trip for input to all four RPS auxiliary trip units. The Loss of Load Trip is actuated by turbine auxiliary relays 305L and 305R. Relay 305L provides input to RPS channels A and C; 305R to channels B and D. Relays 305L and 305R are energized on a turbine trip. Their inputs are the same as the inputs to the turbine solenoid trip valve, 20ET.

  If a turbine trip is generated by loss of auto stop oil pressure, the auto stop oil pressure switches, by two-out-of-three logic, will actuate relays 305L and 305R and generate a reactor trip. If a turbine trip is generated by an input to the solenoid trip valve, relays 305L and 305R, which are wired in parallel, will also be actuated and will generate a reactor trip.

- Containment High Pressure Trip

  The Containment High Pressure Trip is actuated by four pressure switches, one for each RPS channel.

- Zero Power Mode Bypass Automatic Removal

  The Zero Power Bypass allows manually bypassing (i.e., disabling) four reactor trip functions, Low PCS Flow, Low SG A Pressure, Low SG B Pressure, and TM/LP (low PCS pressure), when reactor power (as indicated by the wide range nuclear instrument channels) is below 10^{-4}%. This bypassing is necessary to allow RPS testing and control rod drive mechanism testing when the reactor is shutdown and plant conditions would cause a reactor trip to be present.

  The Zero Power Mode Bypass removal interlock uses the wide range nuclear instruments (NIs) as measurement channels. There are only two wide range NI channels. Separate bistables are provided to actuate the bypass removal for each RPS channel. Bistables in the NI-1/3 channel provide the bypass removal function for RPS channels A and C; bistables in the NI-2/4 channel for RPS channels B and D.

BASES

BACKGROUND (continued)

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.

RPS Trip Units

Two types of RPS trip units are used in the RPS cabinets; bistable trip units and auxiliary trip units:

> A bistable trip unit receives a measured process signal from its instrument channel and compares it to a setpoint; the trip unit actuates three relays, with contacts in the Matrix Logic channels, when the measured signal is less conservative than the setpoint. They also provide local trip indication and remote annunciation.
>
> An auxiliary trip unit receives a digital input (contacts open or closed); the trip unit actuates three relays, with contacts in the Matrix Logic channels, when the digital input is received. They also provide local trip indication and remote annunciation.

Each RPS channel has four auxiliary trip units and seven bistable trip units.

The contacts from these trip unit relays are arranged into six coincidence matrices, comprising the Matrix Logic. If bistable trip units monitoring the same parameter in at least two channels trip, the Matrix Logic will generate a reactor trip (two-out-of-four logic).

Four of the RPS measurement channels provide contact outputs to the RPS, so the comparison of an analog input to a trip setpoint is not necessary. In these cases, the bistable trip unit is replaced with an auxiliary trip unit. The auxiliary trip units provide contact multiplication so the single input contact opening can provide multiple contact outputs to the coincidence logic as well as trip indication and annunciation.

BASES

BACKGROUND (continued)

RPS Trip Units (continued)

Trips employing auxiliary trip units include the VHPT, which receives contact inputs from the Thermal Margin Monitors; the High Startup Rate trip which employs contact inputs from bistables mounted in the two wide range drawers; the Loss of Load Trip which receives contact inputs from one of two auxiliary relays which are operated by two-out-of-three logic switches sensing turbine auto stop oil pressure; and the Containment High Pressure (CHP) trip, which employs containment pressure switch contacts.

There are four RPS trip units, designated as channels A through D, each channel having eleven trip units, one for each RPS Function. Trip unit output relays de-energize when a trip occurs.

All RPS Trip Functions, with the exception of the Loss of Load and CHP trips, generate a pretrip alarm as the trip setpoint is approached.

The Allowable Values are specified for each safety related RPS 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 SLs of Chapter 2.0 are not violated during AOOs and the consequences of 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.

Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the LSSS.

BASES

BACKGROUND (continued)

Reactor Protective System Bypasses

Three different types of trip bypass are utilized in the RPS, Operating Bypass, Zero Power Mode Bypass, and Trip Channel Bypass. The Operating Bypass or Zero Power Mode Bypass prevent the actuation of a trip unit or auxiliary trip unit; the Trip Channel Bypass prevents the trip unit output from affecting the Logic Matrix. A channel which is bypassed, other than as allowed by the Table 3.3.1-1 footnotes, cannot perform its specified safety function and must be considered to be inoperable.

Operating Bypasses

The Operating Bypasses are initiated and removed automatically during startup and shutdown as power level changes. An Operating Bypass prevents the associated RPS auxiliary trip unit from receiving a trip signal from the associated measurement channel. With the bypass in place, neither the pre-trip alarm nor the trip will actuate if the measured parameter exceeds the set point. An annunciator is provided for each Operating Bypass. The RPS trips with Operating Bypasses are:

a. High Startup Rate Trip bypass. The High Startup Rate trip is automatically bypassed when the associated wide range channel indicates below 1E-4% RTP, and when the associated power range excore channel indicates above 13% RTP. These bypasses are automatically removed between 1E-4% RTP and 13% RTP.

b. Loss of Load bypass. The Loss of Load trip is automatically bypassed when the associated power range excore channel indicates below 17% RTP. The bypass is automatically removed when the channel indicates above the set point. The same power range excore channel bistable is used to bypass the High Startup Rate trip and the Loss of Load trip for that RPS channel.

BASES

BACKGROUND (continued)

Operating Bypasses (continued)

Each wide range channel contains two bistables set at 1E-4% RTP, one bistable unit for each associated RPS channel. Each of the two wide range channels affect the Operating Bypasses for two RPS channels; wide range channel NI-1/3 for RPS channels A and C, wide range channel NI-2/4 for RPS channels B and D. Each of the four power range excore channel affects the Operating Bypasses for the associated RPS channel. The power range excore channel bistables associated with the Operating Bypasses are set at a nominal 15%, and are required to actuate between 13% RTP and 17% RTP.

Zero Power Mode (ZPM) Bypass

The ZPM Bypass is used when the plant is shut down and it is desired to raise the control rods for control rod drop testing with PCS flow, pressure or temperature too low for the RPS trips to be reset. ZPM bypasses may be manually initiated and removed when wide range power is below 1E-4% RTP, and are automatically removed if the associated wide range NI indicated power exceeds 1E-4% RTP. A ZPM bypass prevents the RPS trip unit from actuating if the measured parameter exceeds the set point. Operation of the pretrip alarm is unaffected by the zero power mode bypass. An annunciator indicates the presence of any ZPM bypass. The RPS trips with ZPM bypasses are:

a. Low Primary Coolant System Flow.

b. Low Steam Generator Pressure.

c. Thermal Margin/Low Pressure.

The wide range NI channels provide contact closure permissive signals when indicated power is below 1E-4% RTP. The ZPM bypasses may then be manually initiated or removed by actuation of key-lock switches. One key-lock switch located on each RPS cabinet controls the ZPM Bypass for the associated RPS trip channels. The bypass is automatically removed if the associated wide range NI indicated power exceeds 1E-4% RTP. The same wide range NI channel bistables that provide the ZPM Bypass permissive and removal signals also provide the high startup rate trip Operating Bypass actuation and removal.

BASES

BACKGROUND (continued)

Trip Channel Bypass

A Trip Channel Bypass is used when it is desired to physically remove an individual trip unit from the system, or when calibration or servicing of a trip channel could cause an inadvertent trip. A trip Channel Bypass may be manually initiated or removed at any time by actuation of a key-lock switch. A Trip Channel Bypass prevents the trip unit output from affecting the RPS logic matrix. A light above the bypass switch indicates that the trip channel has been bypassed. Each RPS trip unit has an associated trip channel bypass:

The key-lock trip channel bypass switch is located above each trip unit. The key cannot be removed when in the bypass position. Only one key for each trip parameter is provided, therefore the operator can bypass only one channel of a given parameter at a time. During the bypass condition, system logic changes from two-out-of-four to two-out-of-three channels required for trip.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Reference 4 takes credit for most RPS trip Functions. The High Startup Rate and Loss of Load Functions, which are not specifically credited in the accident analysis, are part of the NRC approved licensing basis for the plant, and are required to be operable in accordance with their respective LCO. The High Startup Rate and Loss of Load trips are purely equipment protective, and their use minimizes the potential for equipment damage.

The specific safety analyses applicable to each protective Function are identified below.

1. Variable High Power Trip (VHPT)

   The VHPT provides reactor core protection against positive reactivity excursions.

   The safety analysis assumes that this trip is OPERABLE to terminate excessive positive reactivity insertions during power operation and while shut down.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

2. High Startup Rate Trip

There are no safety analyses which take credit for functioning of the High Startup Rate Trip. The High Startup Rate trip is used to trip the reactor when excore wide range power indicates an excessive rate of change. The High Startup Rate trip minimizes transients for events such as a continuous control rod withdrawal or a boron dilution event from low power levels. The trip may be operationally bypassed when THERMAL POWER is < 1E-4% RTP, when poor counting statistics may lead to erroneous indication. It may also be operationally bypassed at > 13% RTP, where moderator temperature coefficient and fuel temperature coefficient make high rate of change of power unlikely.

There are only two wide range drawers, with each supplying contact input to auxiliary trip units in two RPS channels.

3. Low Primary Coolant System Flow Trip

The Low PCS Flow trip provides DNB protection during events which suddenly reduce the PCS flow rate during power operation, such as loss of power to, or seizure of, a primary coolant pump.

Flow in each of the four PCS loops is determined from pressure drop from inlet to outlet of the SGs. The total PCS flow is determined, for the RPS flow channels, by summing the loop pressure drops across the SGs and correlating this pressure sum with the sum of SG differential pressures which exist at 100% flow (four pump operation at full power T_{ave}). Full PCS flow is that flow which exists at RTP, at full power T_{ave}, with four pumps operating.

4, 5. Low Steam Generator Level Trip

The Low Steam Generator Level trips are provided to trip the reactor in the event of excessive steam demand (to prevent overcooling the PCS) and loss of feedwater events (to prevent overpressurization of the PCS).

The Allowable Value assures that there will be sufficient water inventory in the SG at the time of trip to allow a safe and orderly plant shutdown and to prevent SG dryout assuming minimum AFW capacity.

APPLICABLE SAFETY ANALYSIS (continued)

4, 5. Low Steam Generator Level Trip (continued)

Each SG level is sensed by measuring the differential pressure in the upper portion of the downcomer annulus in the SG. These trips share four level sensing channels on each SG with the AFW actuation signal.

6, 7. Low Steam Generator Pressure Trip

The Low Steam Generator Pressure trip provides protection against an excessive rate of heat extraction from the steam generators, which would result in a rapid uncontrolled cooldown of the PCS. This trip provides a mitigation function in the event of an MSLB.

The Low SG Pressure channels are shared with the Low SG Pressure signals which isolate the steam and feedwater lines.

8. High Pressurizer Pressure Trip

The High Pressurizer Pressure trip, in conjunction with pressurizer safety valves and Main Steam Safety Valves (MSSVs), provides protection against overpressure conditions in the PCS when at operating temperature. The safety analyses assume the High Pressurizer Pressure trip is OPERABLE during accidents and transients which suddenly reduce PCS cooling (e.g., Loss of Load, Main Steam Isolation Valve (MSIV) closure, etc.) or which suddenly increase reactor power (e.g., rod ejection accident).

The High Pressurizer Pressure trip shares four safety grade instrument channels with the TM/LP trip, Anticipated Transient Without Scram (ATWS) and PORV circuits, and the Pressurizer Low Pressure Safety Injection Signal.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

9. Thermal Margin/Low Pressure (TM/LP) Trip

The TM/LP trip is provided to prevent reactor operation when the DNBR is insufficient. The TM/LP trip protects against slow reactivity or temperature increases, and against pressure decreases.

The trip is initiated whenever the PCS pressure signal drops below a minimum value (P_{min}) or a computed value (P_{var}) as described below, whichever is higher.

The TM/LP trip uses Q Power, ASI, pressurizer pressure, and cold leg temperature (T_c) as inputs.

Q Power is the higher of core THERMAL POWER (ΔT Power) or nuclear power. The ΔT power uses hot leg and cold leg RTDs as inputs. Nuclear power uses the power range excore channels as inputs. Both the ΔT and excore power signals have provisions for calibration by calorimetric calculations.

The ASI is calculated from the upper and lower power range excore detector signals, as explained in Section 1.1, "Definitions." The signal is corrected for the difference between the flux at the core periphery and the flux at the detectors.

The T_c value is the higher of the two cold leg signals.

The Low Pressurizer Pressure trip limit (P_{var})is calculated using the equations given in Table 3.3.1-2.

The calculated limit (P_{var}) is then compared to a fixed Low Pressurizer Pressure trip limit (P_{min}). The auctioneered highest of these signals becomes the trip limit (P_{trip}). P_{trip} is compared to the measured PCS pressure and a trip signal is generated when the measured pressure for that channel is less than or equal to P_{trip}. A pre-trip alarm is also generated when P is less than or equal to the pre-trip setting, P_{trip} + ΔP.

The TM/LP trip setpoint is a complex function of these inputs and represents a minimum acceptable PCS pressure for the existing temperature and power conditions. It is compared to actual PCS pressure in the TM/LP trip unit.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

10. Loss of Load Trip

There are no safety analyses which take credit for functioning of the Loss of Load Trip.

The Loss of Load trip is provided to prevent lifting the pressurizer and main steam safety valves in the event of a turbine generator trip while at power. The trip is equipment protective. The safety analyses do not assume that this trip functions during any accident or transient. The Loss of Load trip uses two-out-of-three logic from pressure switches in the turbine auto stop oil circuit to sense a turbine trip for input to all four RPS auxiliary trip units.

11. Containment High Pressure Trip

The Containment High Pressure trip provides a reactor trip in the event of a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The Containment High Pressure trip shares sensors with the Containment High Pressure sensing logic for Safety Injection, Containment Isolation, and Containment Spray. Each of these sensors has a single bellows which actuates two microswitches. One microswitch on each of four sensors provides an input to the RPS.

12. Zero Power Mode Bypass Removal

The only RPS bypass considered in the safety analyses is the Zero Power Mode (ZPM) Bypass. The ZPM Bypass is used when the plant is shut down and it is desired to raise the control rods for control rod drop testing with PCS flow or temperature too low for the RPS Low PCS Flow, Low SG Pressure, or Thermal Margin/Low Pressure trips to be reset. ZPM bypasses are automatically removed if the wide range NI indicated power exceeds 1E-4% RTP.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

12. Zero Power Mode Bypass Removal (continued)

The safety analyses take credit for automatic removal of the ZPM Bypass if reactor criticality due to a Continuous Control Rod Bank Withdrawal should occur with the affected trips bypassed and PCS flow, pressure, or temperature below the values at which the RPS could be reset. The ZPM Bypass would effectively be removed when the first wide range NI channel indication reached 1E-4% RTP. With the ZPM Bypass for two RPS channels removed, the RPS would trip on one of the un-bypassed trips. This would prevent the reactor reaching an excessive power level.

If a reactor criticality due to a Continuous Control Rod Bank Withdrawal should occur when PCS flow, steam generator pressure, and PCS pressure (TM/LP) were above their trip setpoints, a trip would terminate the event when power increased to the minimum setting (nominally 30%) of the Variable High Power Trip. In this case, the monitored parameters are at or near their normal operational values, and a trip initiated at 30% RTP provides adequate protection.

The RPS design also includes automatic removal of the Operating Bypasses for the High Startup Rate and Loss of Load trips. The safety analyses do not assume functioning of either these trips or the automatic removal of their bypasses.

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

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of the trip unit (including its output relays), any required portion of the associated instrument channel, or both, renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. Failure of an automatic ZPM bypass removal channel may also impact the associated instrument channel(s) and reduce the reliability of the affected Functions.

BASES

LCO
(continued)

Actions allow Trip Channel Bypass of individual channels, but the bypassed channel must be considered to be inoperable. The bypass key used to bypass a single channel cannot be simultaneously used to bypass that same parameter in other channels. This interlock prevents operation with more than one channel of the same Function trip channel bypassed. The plant is normally restricted to 7 days in a trip channel bypass, or otherwise inoperable condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic).

The Allowable Values are specified for each safety related RPS 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. 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. Neither Allowable Values nor setpoints are specified for the non-safety related RPS Trip Functions, since no safety analysis assumptions would be violated if they are not set at a particular value.

The following Bases for each trip Function identify the above RPS trip Function criteria items that are applicable to establish the trip Function OPERABILITY.

1. Variable High Power Trip (VHPT)

This LCO requires all four channels of the VHPT Function to be OPERABLE.

The Allowable Value is high enough to provide an operating envelope that prevents unnecessary VHPT trips during normal plant operations. The Allowable Value is low enough for the system to function adequately during reactivity addition events.

BASES

LCO (continued)

1. Variable High Power Trip (VHPT) (continued)

The VHPT is designed to limit maximum reactor power to its maximum design and to terminate power excursions initiating at lower powers without power reaching this full power limit. During plant startup, the VHPT trip setpoint is initially at its minimum value, ≤ 30%. Below 30% RTP, the VHPT setpoint is not required to “track” with Q Power, i.e., be adjusted to within 15% RTP. It remains fixed until manually reset, at which point it increases to ≤ 15% above existing Q Power.

The maximum allowable setting of the VHPT is 109.4% RTP. Adding to this the possible variation in trip setpoint due to calibration and instrument error, the maximum actual steady state power at which a trip would be actuated is 113.4%, which is the value assumed in the safety analysis.

2. High Startup Rate Trip

This LCO requires four channels of High Startup Rate Trip Function to be OPERABLE in MODES 1 and 2.

The High Startup Rate trip serves as a backup to the administratively enforced startup rate limit. The Function is not credited in the accident analyses; therefore, no Allowable Value for the trip or operating bypass Functions is derived from analytical limits and none is specified.

The High Startup Rate Trip is required to be OPERABLE, in accordance with the LCO, even though the Trip Function is not credited in the accident analysis.

The four channels of the High Startup Rate trip are derived from two wide range NI signal processing drawers. Thus, a failure in one wide range channel could render two RPS channels inoperable. It is acceptable to continue operation in this condition because the High Startup Rate trip is not credited in any safety analyses.

The requirement for this trip Function is modified by a footnote, which allows the High Startup Rate trip to be bypassed when the wide range NI indicates below 10E-4% or when THERMAL POWER is above 13% RTP. If a High Startup Rate trip is bypassed when power is between these limits, it must be considered to be inoperable.

BASES

LCO (continued)

3. Low Primary Coolant System Flow Trip

This LCO requires four channels of Low PCS Flow Trip Function to be OPERABLE.

This trip is set high enough to maintain fuel integrity during a loss of flow condition. The setting is low enough to allow for normal operating fluctuations from offsite power.

The Low PCS Flow trip setpoint of 95% of full PCS flow insures that the reactor cannot operate when the flow rate is less than 93% of the nominal value considering instrument errors. Full PCS flow is that flow which exists at RTP, at full power Tave, with four pumps operating.

The requirement for this trip Function is modified by a footnote, which allows use of the ZPM bypass when wide range power is below 1E-4% RTP. That bypass is automatically removed when the associated wide range channel indicates 1E-4% RTP. If a trip channel is bypassed when power is above 1E-4% RTP, it must be considered to be inoperable.

4, 5. Low Steam Generator Level Trip

This LCO requires four channels of Low Steam Generator Level Trip Function per steam generator to be OPERABLE.

The 25.9% Allowable Value assures that there is an adequate water inventory in the steam generators when the reactor is critical and is based upon narrow range instrumentation. The 25.9% indicated level corresponds to the location of the feed ring.

6, 7. Low Steam Generator Pressure Trip

This LCO requires four channels of Low Steam Generator Pressure Trip Function per steam generator to be OPERABLE.

The Allowable Value of 500 psia is sufficiently below the full load operating value for steam pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of excessive steam demand. Since excessive steam demand causes the PCS to cool down, resulting in positive reactivity addition to the core, a reactor trip is required to offset that effect.

BASES

LCO (continued)

8. <u>High Pressurizer Pressure Trip</u>

This LCO requires four channels of High Pressurizer Pressure Trip Function to be OPERABLE.

The Allowable Value is set high enough to allow for pressure increases in the PCS during normal operation (i.e., plant transients) not indicative of an abnormal condition. The setting is below the lift setpoint of the pressurizer safety valves and low enough to initiate a reactor trip when an abnormal condition is indicated.

9. <u>Thermal Margin/Low Pressure (TM/LP) Trip</u>

This LCO requires four channels of TM/LP Trip Function to be OPERABLE.

The TM/LP trip setpoints are derived from the core thermal limits through application of appropriate allowances for measurement uncertainties and processing errors. The allowances specifically account for instrument drift in both power and inlet temperatures, calorimetric power measurement, inlet temperature measurement, and primary system pressure measurement.

Other uncertainties including allowances for assembly power tilt, fuel pellet manufacturing tolerances, core flow measurement uncertainty and core bypass flow, inlet temperature measurement time delays, and ASI measurement, are included in the development of the TM/LP trip setpoint used in the accident analysis.

The requirement for this trip Function is modified by a footnote, which allows use of the ZPM bypass when wide range power is below 1E-4% RTP. That bypass is automatically removed when the associated wide range channel indicates 1E-4% RTP. If a trip channel is bypassed when power is above 1E-4% RTP, it must be considered to be inoperable.

BASES

LCO (continued)

10. <u>Loss of Load Trip</u>

The LCO requires four Loss of Load Trip Function channels to be OPERABLE in MODE 1 with THERMAL POWER ≥ 17% RTP.

The Loss of Load trip may be bypassed or be inoperable with THERMAL POWER < 17% RTP, since it is no longer needed to prevent lifting of the pressurizer safety valves or steam generator safety valves in the event of a Loss of Load. Loss of Load Trip unit must be considered inoperable if it is bypassed when THERMAL POWER is above 17% RTP.

This LCO requires four RPS Loss of Load auxiliary trip units, relays 305L and 305R, and pressure switches 63/AST-1, 63/AST-2, and 63/AST-3 to be OPERABLE. With those components OPERABLE, a turbine trip will generate a reactor trip. The LCO does not require the various turbine trips, themselves, to be OPERABLE.

The Nuclear Steam Supply System and Steam Dump System are capable of accommodating the Loss of Load without requiring the use of the above equipment.

The Loss of Load Trip Function is not credited in the accident analysis; therefore, an Allowable Value for the trip cannot be derived from analytical limits, and is not specified.

The Loss of Load Trip is required to be OPERABLE, in accordance with the LCO, even though the Trip Function is not credited in the accident analysis.

11. <u>Containment High Pressure Trip</u>

This LCO requires four channels of Containment High Pressure Trip Function to be OPERABLE.

The Allowable Value is high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) that are not indicative of an abnormal condition. The setting is low enough to initiate a reactor trip to prevent containment pressure from exceeding design pressure following a DBA and ensures the reactor is shutdown before initiation of safety injection and containment spray.

BASES

LCO (continued)

12. ZPM Bypass

The LCO requires that four channels of automatic Zero Power Mode (ZPM) Bypass removal instrumentation be OPERABLE. Each channel of automatic ZPM Bypass removal includes a shared wide range NI channel, an actuating bistable in the wide range drawer, and a relay in the associated RPS cabinet. Wide Range NI channel 1/3 is shared between ZPM Bypass removal channels A and C; Wide Range NI channel 2/4, between ZPM Bypass removal channels B and D. An operable bypass removal channel must be capable of automatically removing the capability to bypass the affected RPS trip channels with the ZPM Bypass key switch at the proper setpoint.

APPLICABILITY

This LCO requires all safety related trip functions to be OPERABLE in accordance with Table 3.3.1-1.

Those RPS trip Functions which are assumed in the safety analyses (all except High Startup Rate and Loss of Load), are required to be operable in MODES 1 and 2, and in MODES 3, 4, and 5 with more than one full-length control rod capable of being withdrawn and PCS boron concentration less than REFUELING BORON CONCENTRATION.

These trip Functions are not required while in MODES 3, 4, or 5, if PCS boron concentration is at REFUELING BORON CONCENTRATION, or when no more than one full-length control rod is capable of being withdrawn, because the RPS Function is already fulfilled. REFUELING BORON CONCENTRATION provides sufficient negative reactivity to assure the reactor remains subcritical regardless of control rod position, and the safety analyses assume that the highest worth withdrawn full-length control rod will fail to insert on a trip. Therefore, under these conditions, the safety analyses assumptions will be met without the RPS trip Function.

The High Startup Rate Trip Function is required to be OPERABLE in MODES 1 and 2, but may be bypassed when the associated wide range NI channel indicates below 1E-4% power, when poor counting statistics may lead to erroneous indication. In MODES 3, 4, 5, and 6, the High Startup Rate trip is not required to be OPERABLE. Wide range channels are required to be OPERABLE in MODES 3, 4, and 5, by LCO 3.3.9, "Neutron Flux Monitoring Channels," and in MODE 6, by LCO 3.9.2, “Nuclear Instrumentation.”

BASES

APPLICABILITY (continued)

The High Startup Rate Trip Function is required to be OPERABLE in MODES 1 and 2, but may be bypassed when the associated wide range NI channel indicates below 1E-4% power, when poor counting statistics may lead to erroneous indication. In MODES 3, 4, 5, and 6, the High Startup Rate trip is not required to be OPERABLE. Wide range channels are required to be OPERABLE in MODES 3, 4, and 5, by LCO 3.3.9, "Neutron Flux Monitoring Channels," and in MODE 6, by LCO 3.9.2, “Nuclear Instrumentation.”

The Loss of Load trip is required to be OPERABLE with THERMAL POWER at or above 17% RTP. Below 17% RTP, the ADVs are capable of relieving the pressure due to a Loss of Load event without challenging other overpressure protection.

The trips are designed to take the reactor subcritical, maintaining the SLs during AOOs and assisting the ESF in providing acceptable consequences during accidents.

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).

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or RPS bistable trip unit is found inoperable, all affected Functions provided by that channel must be declared inoperable, and the plant must enter the Condition for the particular protection Functions affected.

BASES

ACTIONS (continued)

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

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Times of each inoperable Function will be tracked separately for each Function, starting from the time the Condition was entered.

A.1

Condition A applies to the failure of a single channel in any required RPS Function, except High Startup Rate, Loss of Load, or ZPM Bypass Removal. (Condition A is modified by a Note stating that this Condition does not apply to the High Startup Rate, Loss of Load, or ZPM Bypass Removal Functions. The failure of one channel of those Functions is addressed by Conditions B, C, or D.)

If one RPS bistable trip unit or associated instrument channel is inoperable, operation is allowed to continue. Since the trip unit and associated instrument channel combine to perform the trip function, this Condition is also appropriate if both the trip unit and the associated instrument channel are inoperable. Though not required, the inoperable channel may be bypassed. The provision of four trip channels allows one channel to be bypassed (removed from service) during operations, placing the RPS in two-out-of-three coincidence logic. The failed channel must be restored to OPERABLE status or placed in trip within 7 days.

Required Action A.1 places the Function in a one-out-of-three configuration. In this configuration, common cause failure of dependent channels cannot prevent trip.

The Completion Time of 7 days is based on 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)

A.1 (continued)

The Completion Time of 7 days is based on operating experience, which has demonstrated that a random failure of a second channel occurring during the 7 day period is a low probability event.

B.1

Condition B applies to the failure of a single High Startup Rate trip unit or associated instrument channel.

If one trip unit or associated instrument channel fails, it must be restored to OPERABLE status prior to entering MODE 2 from MODE 3. A shutdown provides the appropriate opportunity to repair the trip function and conduct the necessary testing. The Completion Time is based on the fact that the safety analyses take no credit for the functioning of this trip.

C.1

Condition C applies to the failure of a single Loss of Load or associated instrument channel.

If one trip unit or associated instrument channel fails, it must be restored to OPERABLE status prior to THERMAL POWER ≥ 17% RTP following a shutdown. If the plant is shutdown at the time the channel becomes inoperable, then the failed channel must be restored to OPERABLE status prior to THERMAL POWER ≥ 17% RTP. For this Completion Time, “following a shutdown” means this Required Action does not have to be completed until prior to THERMAL POWER ≥ 17% RTP for the first time after the plant has been in MODE 3 following entry into the Condition. The Completion Time trip assures that the plant will not be restarted with an inoperable Loss of Load trip channel.

BASES

ACTIONS (continued)

D.1 and D.2

Condition D applies when one or more automatic ZPM Bypass removal channels are inoperable. If the ZPM Bypass removal channel cannot be restored to OPERABLE status, the affected ZPM Bypasses must be immediately removed, or the bypassed RPS trip Function channels must be immediately declared to be inoperable. Unless additional circuit failures exist, the ZPM Bypass may be removed by placing the associated “Zero Power Mode Bypass” key operated switch in the normal position.

A trip channel which is actually bypassed, other than as allowed by the Table 3.3.1-1 footnotes, cannot perform its specified safety function and must immediately be declared to be inoperable.

E.1 and E.2

Condition E applies to the failure of two channels in any RPS Function, except ZPM Bypass Removal Function. (The failure of ZPM Bypass Removal Functions is addressed by Condition D.).

Condition E is modified by a Note stating that this Condition does not apply to the ZPM Bypass Removal Function.

Required Action E.1 provides for placing one inoperable channel in trip within the Completion Time of 1 hour. Though not required, the other inoperable channel may be (trip channel) bypassed.

ACTIONS (continued)

E.1 and E.2 (continued)

This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels while ensuring that the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed or inoperable in an untripped condition, the RPS is in a two-out-of-three logic for that function; but with another channel failed, the RPS may be operating in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, one of the inoperable channels is placed in trip. This places the RPS in a one-out-of-two for that function logic. If any of the other unbypassed channels for that function receives a trip signal, the reactor will trip.

Action E.2 is modified by a Note stating that this Action does not apply to (is not required for) the High Startup Rate and Loss of Load Functions.

One channel is required to be restored to OPERABLE status within 7 days for reasons similar to those stated under Condition A. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action E.2 must be placed in trip if more than 7 days have elapsed since the initial channel failure.

F.1

The power range excore channels are used to generate the internal ASI signal used as an input to the TM/LP trip. They also provide input to the Thermal Margin Monitors for determination of the Q Power input for the TM/LP trip and the VHPT. If two power range excore channels cannot be restored to OPERABLE status, power is restricted or reduced during subsequent operations because of increased uncertainty associated with inoperable power range excore channels which provide input to those trips.

The Completion Time of 2 hours is adequate to reduce power in an orderly manner without challenging plant systems.

BASES

ACTIONS (continued)

G.1, G.2.1, and G.2.2

Condition G is entered when the Required Action and associated Completion Time of Condition A, B, C, D, E, or F are not met, or if the control room ambient air temperature exceeds 90°F.

If the control room ambient air temperature exceeds 90°F, all Thermal Margin Monitor channels are rendered inoperable because their operating temperature limit is exceeded. In this condition, or if the Required Actions and associated Completion Times are not met, the reactor must be placed in a condition in which the LCO does not apply. To accomplish this, the plant must be placed in MODE 3, with no more than one full-length control rod capable of being withdrawn or with the PCS boron concentration at REFUELING BORON CONCENTRATION in 6 hours.

The Completion Time is reasonable, based on operating experience, for placing the plant in MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The Completion Time is also reasonable to ensure that no more than one full-length control rod is capable of being withdrawn or that the PCS boron concentration is at REFUELING BORON CONCENTRATION.

SURVEILLANCE REQUIREMENTS

The SRs for any particular RPS Function are found in the SR column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.1.1

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1 (continued)

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

The Containment High Pressure and Loss of Load channels are pressure switch actuated. As such, they have no associated control room indicator and do not require a CHANNEL CHECK.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.2

This SR verifies that the control room ambient air temperature is within the environmental qualification temperature limits for the most restrictive RPS components, which are the Thermal Margin Monitors. These monitors provide input to both the VHPT Function and the TM/LP Trip Function. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.3

A calibration (heat balance) is performed when THERMAL POWER is ≥ 15%. The calibration consists of adjusting the "nuclear power calibrate" potentiometers to agree with the calorimetric calculation if the absolute difference is ≥ 1.5%. Nuclear power is adjusted via a potentiometer, or THERMAL POWER is adjusted via a Thermal Margin Monitor bias number, as necessary, in accordance with the calibration (heat balance) procedure. Performance of the calibration ensures that the two inputs to the Q power measurement are indicating accurately with respect to the much more accurate secondary calorimetric calculation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The Frequency is modified by a Note indicating this Surveillance must be performed within 12 hours after THERMAL POWER is ≥ 15% RTP. The secondary calorimetric is inaccurate at lower power levels. The 12 hours allows time requirements for plant stabilization, data taking, and instrument calibration.

SR 3.3.1.4

It is necessary to calibrate the power range excore channel upper and lower subchannel amplifiers such that the measured ASI reflects the true core power distribution as determined by the incore detectors. ASI is utilized as an input to the TM/LP trip function where it is used to ensure that the measured axial power profiles are bounded by the axial power profiles used in the development of the T_{inlet} limitation of LCO 3.4.1. An adjustment of the excore channel is necessary only if reactor power is greater than 25% RTP and individual excore channel ASI differs from AXIAL OFFSET, as measured by the incores, outside the bounds of the following table:

Allowed Reactor Power	Group 4 Rods ≥ 128" withdrawn	Group 4 Rods <128" withdrawn
≤ 100%	-0.020 ≤ (AO-ASI) ≤ 0.020	-0.040 ≤ (AO-ASI) ≤ 0.040
< 95	-0.033 ≤ (AO-ASI) ≤ 0.020	-0.053 ≤ (AO-ASI) ≤ 0.040
< 90	-0.046 ≤ (AO-ASI) ≤ 0.020	-0.066 ≤ (AO-ASI) ≤ 0.040
< 85	-0.060 ≤ (AO-ASI) ≤ 0.020	-0.080 ≤ (AO-ASI) ≤ 0.040
< 80	-0.120 ≤ (AO-ASI) ≤ 0.080	-0.140 ≤ (AO-ASI) ≤ 0.100
< 75	-0.120 ≤ (AO-ASI) ≤ 0.080	-0.140 ≤ (AO-ASI) ≤ 0.100
< 70	-0.120 ≤ (AO-ASI) ≤ 0.080	-0.140 ≤ (AO-ASI) ≤ 0.100
< 65	-0.120 ≤ (AO-ASI) ≤ 0.080	-0.140 ≤ (AO-ASI) ≤ 0.100
< 60	-0.160 ≤ (AO-ASI) ≤ 0.120	-0.180 ≤ (AO-ASI) ≤ 0.140
< 55	-0.160 ≤ (AO-ASI) ≤ 0.120	-0.180 ≤ (AO-ASI) ≤ 0.140
< 50	-0.160 ≤ (AO-ASI) ≤ 0.120	-0.180 ≤ (AO-ASI) ≤ 0.140
< 45	-0.160 ≤ (AO-ASI) ≤ 0.120	-0.180 ≤ (AO-ASI) ≤ 0.140
< 40	-0.160 ≤ (AO-ASI) ≤ 0.120	-0.180 ≤ (AO-ASI) ≤ 0.140
< 35	-0.160 ≤ (AO-ASI) ≤ 0.120	-0.180 ≤ (AO-ASI) ≤ 0.140
< 30	-0.160 ≤ (AO-ASI) ≤ 0.120	-0.180 ≤ (AO-ASI) ≤ 0.140
< 25	Below 25% RTP any AO/ASI difference is acceptable	

Table values determined with a conservative P_{var} gamma constant of –9505.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4 (continued)

Below 25% RTP any difference between ASI and AXIAL OFFSET is acceptable. A Note indicates the Surveillance is not required to have been performed until 12 hours after THERMAL POWER is ≥ 25% RTP. Uncertainties in the excore and incore measurement process make it impractical to calibrate when THERMAL POWER is < 25% RTP. The 12 hours allows time for plant stabilization, data taking, and instrument calibration.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.5

A CHANNEL FUNCTIONAL TEST is performed on each RPS instrument channel, except Loss of Load and High Startup Rate, to ensure the entire channel will perform its intended function when needed. For the TM/LP Function, the constants associated with the Thermal Margin Monitors must be verified to be within tolerances.

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

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

A calibration check of the power range excore channels is performed using the internal test circuitry. This SR uses an internally generated test signal to check that the 0% and 50% levels read within limits for both the upper and lower detector, both on the analog meter and on the TMM screen. This check verifies that neither the zero point nor the amplifier gain adjustment have undergone excessive drift since the previous complete CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.7

A CHANNEL FUNCTIONAL TEST on the Loss of Load and High Startup Rate channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function.

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests.

The High Startup Rate trip is actuated by either of the Wide Range Nuclear Instrument Startup Rate channels. NI-1/3 sends a trip signal to RPS channels A and C; NI-2/4 to channels B and D. Since each High Startup Rate channel would cause a trip on two RPS channels, the High Startup Rate trip is not tested when the reactor is critical.

The four Loss of Load Trip channels are all actuated by a single pressure switch monitoring turbine auto stop oil pressure which is not tested when the reactor is critical. Operating experience has shown that these components usually pass the Surveillance when performed at a Frequency of once per 7 days prior to each reactor startup.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.8

SR 3.3.1.8 performs a CHANNEL CALIBRATION.

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

The bistable setpoints must be found to trip within the Allowable Values specified in the LCO and left set consistent with the assumptions of the setpoint analysis. The Variable High Power Trip setpoint shall be verified to reset properly at several indicated power levels during (simulated) power increases and power decreases.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the setpoint analysis.

As part of the CHANNEL CALIBRATION of the wide range Nuclear Instrumentation, automatic removal of the ZPM Bypass for the Low PCS Flow, TM/LP must be verified to assure that these trips are available when required.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note which states that it is not necessary to calibrate neutron detectors because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in power range excore neutron detector sensitivity are compensated for by performing the calorimetric calibration (SR 3.3.1.3) and the calibration using the incore detectors (SR 3.3.1.4). Sudden changes in detector performance would be noted during the required CHANNEL CHECKS (SR 3.3.1.1).

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 21
2. 10 CFR 100
3. IEEE Standard 279-1971, April 5, 1972
4. FSAR, Chapter 14
5. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989
6. 10 CFR 50.67

Table B 3.3.1-1 (page 1 of 1)
Instruments Affecting Multiple Specifications

Required Instrument Channels	Affected Specifications
Nuclear Instrumentation	
Source Range NI-1/3, Count Rate Indication @ C-150 Panel	3.3.8 (#1)
Source Range NI-1/3 & 2/4, Count Rate Signal	3.3.9 & 3.9.2
Wide Range NI-1/3 & 2/4, Flux Level 10^{-4} Bypass	3.3.1 (#3, 6, 7, 9, & 12)
Wide Range NI-1/3 & 2/4, Startup Rate	3.3.1 (#2)
Wide Range NI-1/3 & 2/4, Flux Level Indication @EC-06 Panel for 3.3.7	3.3.7 (#3) & 3.3.9
Power Range NI-5, 6, 7, & 8, Tq	3.2.1 & 3.2.3
Power Range NI-5, 6, 7, & 8, Q Power	3.3.1 (#1 & 9)
Power Range NI-5, 6, 7, & 8, ASI	3.3.1 (#9) & 3.2.1 & 3.2.4
Power Range NI-5, 6, 7, & 8, Loss of Load/High Startup Rate Bypass	3.3.1 (#2 & 10)
PCS T-Cold Instruments	
TT-0112CA, Temperature Signal (SPI ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112CA & 0122CA, Temperature Signal (C-150)	3.3.8 (#6 & 7)
TT-0122CB, Temperature Signal (PIP ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112CA & 0122CB, Temperature Signal (LTOP)	3.4.12.b.1
TT-0112CC & 0122CD (PTR-0112 & 0122) Temperature Indication	3.3.7 (#2)
TT-0112 & 0122 CC & CD, Temperature Signal (SMM)	3.3.7 (#5)
TT-0112 & 0122 CA, CB, CC, & CD, Temperature Signal (Q Power & TMM)	3.3.1 (#1 & 9) & 3.4.1.b
PCS T-Hot Instruments	
TT-0112HA, Temperature Signal (SPI ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112HA & 0122HA, Temperature Signal (C-150)	3.3.8 (#4 & 5)
TT-0122HB, Temperature Signal (PIP ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112 & 0122 HC & HD, Temperature Signal (SMM)	3.3.7 (#5)
TT-0112HC & 0122HD (PTR-0112 & 0122) Temperature Indication	3.3.7 (#1)
TT-0112 & 0122 HA, HB, HC, & HD, Temperature Signal (Q Power & TMM)	3.3.1 (#1 & 9)
Thermal Margin Monitors	
PY-0102A, B, C, & D	3.3.1 (#1 & 9)
Pressurizer Pressure Instruments	
PT-0102A, B, C, & D, Pressure Signal (RPS & SIS)	3.3.1 (#8 & 9) & 3.3.3 (#1.a & 7a)
PT-0104A & B, Pressure Signal (LTOP & SDC Interlock)	3.4.12.b.1 & 3.4.14
PT-0105A & B, Pressure Signal (WR Indication & LTOP)	3.3.7 (#5) & 3.4.12.b.1
PI-0110, Pressure Indication @ C-150 Panel	3.3.8 (#2)
SG Level Instruments	
LT-0751 & 0752 A, B, C, & D, Level Signal (RPS & AFAS)	3.3.1 (#4 & 5) & 3.3.3 (#4.a & 4.b)
LI-0757 & 0758 A & B, Wide Range Level Indication	3.3.7 (#11 & 12)
LI-0757C & 0758C, Wide Range Level Indication @ C-150 Panel	3.3.8 (#10 & 11)
SG Pressure Instruments	
PT-0751 & 0752 A, B, C, & D, Pressure Signal (RPS & SG Isolation) PT-0751A and PT-0752A Pressure Signal (C-150/150A)	3.3.1 (#6 & 7) & 3.3.3 (#2a, 2b, 7b, 7c) 3.3.8 (#8 & 9)
PIC-0751 & 0752 C & D, Pressure Indication	3.3.7 (#13 & 14)
PI-0751E & 0752E, Pressure Indication @ C-150 Panel	3.3.8 (#8 & 9)
Containment Pressure Instruments	
PS-1801, 1802, 1803, & 1804, Switch Output (RPS)	3.3.1 (#11)
PS-1801, 1802A, 1803, & 1804A, Switch Output (ESF)	3.3.3 (#5.a)
PS-1801A, 1802, 1803A, & 1804, Switch Output (ESF)	3.3.3 (#5.b)

Note: The information provided in this table is intended for use as an aid to distinguish those instrument channels which provide more than one required function and to describe which specifications they affect. The information in this table should not be taken as inclusive for all instruments nor affected specifications.

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protective System (RPS) Logic and Trip Initiation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the acceptable fuel design limits and reactor coolant pressure boundary integrity during Anticipated Operational Occurrences (AOOs). (As defined in 10 CFR 50, Appendix A, "Anticipated operational occurrences 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.") By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The Primary Coolant System (PCS) pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within applicable 10 CFR 50.67 (Ref. 6) and 10 CFR 100 (Ref. 2) criteria during AOOs.

BASES

BACKGROUND (continued)

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within applicable 10 CFR 50.67 (Ref. 6) and 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels (or pressure switches);
- Bistable trip units;
- Matrix Logic; and
- Trip Initiation Logic.

This LCO addresses the RPS Logic (Matrix Logic and Trip Initiation Logic), including Manual Trip capability. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," provides a description of the role of the measurement channels and associated bistable trip units in the RPS. The RPS Logic is summarized below:

RPS Logic

The RPS Logic, consisting of Matrix Logic and Trip Initiation Logic, employs a scheme that provides a reactor trip when trip units in any two of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic. This logic and the clutch power supply configuration are shown in FSAR Figure 7-1 (Ref. 3).

Bistable trip unit relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two trip unit channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable trip unit channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relay coils de-energize.

BASES

BACKGROUND (continued)

RPS Logic (continued)

The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized clutch power supply “M-contactors” (M1, M2, M3, and M4). The trip paths thus each have six contacts in series, one from each matrix, and perform a logical OR function, de-energizing the M-contactors if any one or more of the six logic matrices indicate a coincidence condition.

When a coincidence occurs in two RPS channels, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four M-contactors, which interrupt AC input power to the four clutch power supplies, allowing the full-length control rods to insert by gravity.

Manual reactor trip capability is afforded by two main control panel-mounted pushbuttons. One of these (on Control Panel CO-2) opens contacts in series with each of the four trip paths, de-energizing all M-contactors. The other pushbutton (on Control Panel CO-6) opens circuit breakers which provide AC input power to the M-contactor contacts and downstream clutch power supplies. Thus depressing either pushbutton will cause a reactor trip.

De-energizing the M-contactors removes AC power to the four clutch power supply inputs. Contacts from M-contactors M1 and M2 are in series with each other and in the AC power supply path to clutch power supplies PS1 and PS2 (these constitute a “trip leg”). M3 and M4 are similarly arranged with respect to clutch power supplies PS3 and PS4 (these constitute a second “trip leg”). Approximately half of the control rod clutches receive power from auctioneered clutch power supplies 1 and 3. The remaining control rod clutches receive clutch power from auctioneered clutch power supplies 2 and 4.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting RPS cabinet matrix wiring between bistable and auxiliary trip unit relay contacts, including the matrix relays. Contacts in the bistable and auxiliary trip units are excluded from the Matrix Logic definition, since they are addressed as part of the instrumentation channel.

The Trip Initiation Logic consists of the M-contactor isolation transformers, all interconnecting wiring, and the M-contactors.

BASES

BACKGROUND (continued)

RPS Logic (continued)

Manual trip circuitry includes both manual reactor trip pushbuttons C0-2 and C0-6, and the interconnecting wiring necessary to effect deenergization of the clutch power supplies.

Neither the clutch power supplies nor the AC input power source to these supplies is considered as safety related. Operation may continue with one or two selective clutch power supplies de-energized.

It is possible to change the two-out-of-four RPS Logic to a two-out-of-three logic for a given input parameter in one channel at a time by Trip Channel Bypassing the RPS Trip unit output contacts in the Matrix Logic "Ladder." Trip Channel Bypassing a trip unit effectively shorts the trip unit relay contacts in the three matrices associated with that channel. Thus, the bypassed trip units will function normally, producing normal channel trip indication and annunciation, but a reactor trip will not occur unless two additional channels indicate a trip condition. Trip Channel Bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. A single bypass key for each trip function interlock prevents simultaneous Trip Channel Bypassing of the same parameter in more than one channel. Trip Channel Bypassing is normally employed during maintenance or testing.

Functional testing of the entire RPS, from trip unit input through the de-energizing of individual sets of clutch power supplies, can be performed either at power or during shutdown and is normally performed on a quarterly basis. FSAR Section 7.2 (Ref. 4) explains RPS testing in more detail.

APPLICABLE SAFETY ANALYSES

Reactor Protective System (RPS) Logic

The RPS Logic provides for automatic trip initiation to avoid exceeding the SLs during AOOs and to assist the ESF systems in ensuring acceptable consequences during accidents. All transients and accidents that call for a reactor trip assume the RPS Logic is functioning as designed.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

Manual Trip

There are no accident analyses that take credit for the Manual Trip; however, the Manual Trip is part of the RPS circuitry. It is used by the operator to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint. A Manual Trip accomplishes the same results as any one of the automatic trip Functions.

The RPS Logic and Trip Initiation satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

Reactor Protective System (RPS) Logic

Failures of individual trip unit relays and their contacts are addressed in LCO 3.3.1. This Specification addresses failures of the Matrix Logic not addressed in the above, such as the failure of matrix relay power supplies or the failure of the trip channel bypass contact in the bypass condition.

Loss of a single preferred AC bus will de-energize one of the two power supplies in each of three matrices. Because of power supply auctioneering, all four matrix relays will remain energized in each affected matrix.

Each of the four Trip Initiation Logic channels de-energizes one set of clutch power supplies if any of the six coincidence matrices de-energize their associated matrix relays. They thus perform a logical OR function. Trip Initiation Logic channels 1 and 2 receive AC power from preferred AC bus Y-30. Trip Initiation Logic channels 3 and 4 receive AC input power from preferred AC bus Y-40. Because of clutch power supply output auctioneering, it is possible to de-energize either input bus without de-energizing control rod clutches.

1. Matrix Logic

   This LCO requires six channels of Matrix Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one full-length control rod is capable of being withdrawn and the PCS boron concentration is less than REFUELING BORON CONCENTRATION.

BASES

LCO (continued)

2. Trip Initiation Logic

This LCO requires four channels of Trip Initiation Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one full-length control rod is capable of being withdrawn and the PCS boron concentration is less than REFUELING BORON CONCENTRATION.

3. Manual Trip

The LCO requires both Manual Trip channels to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one full-length control rod is capable of being withdrawn and the PCS boron concentration is less than REFUELING BORON CONCENTRATION.

Two independent pushbuttons are provided. Each pushbutton is considered to be a channel. Depressing either pushbutton interrupts power to all four clutch power supplies, tripping the reactor.

APPLICABILITY

The RPS Matrix Logic, Trip Initiation Logic, and Manual Trip are required to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one full-length control rod capable of being withdrawn and the PCS boron concentration is less than REFUELING BORON CONCENTRATION. This ensures the reactor can be tripped when necessary, but allows for maintenance and testing when the reactor trip is not needed.

In MODES 3, 4, and 5 with no more than one full-length control rod capable of being withdrawn or the PCS boron concentration at REFUELING BORON CONCENTRATION, these Functions do not have to be OPERABLE. However, LCO 3.3.9, “Neutron Flux Monitoring Channels,” does require neutron flux monitoring capability under these conditions.

BASES

ACTIONS

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

A.1

Condition A applies if one Matrix Logic channel is inoperable. The channel must be restored to OPERABLE status within 48 hours. The Completion Time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second Matrix Logic channel is low during any given 48 hour interval. If the channel cannot be restored to OPERABLE status within 48 hours, Condition E is entered.

B.1

Condition B applies if one Trip Initiation Logic channel is inoperable. The Required Action require de-energizing the affected clutch power supplies. This removes the need for the affected channel by performing its associated safety function. With the clutch power supplies associated with one initiation logic channel de-energized, the remaining two clutch power supplies prevent control rod clutches from de-energizing. The remaining clutch power supplies are in a one-out-of-two logic with respect to the remaining initiation logic channels in the clutch power supply path. This meets redundancy requirements, but testing on the OPERABLE channels cannot be performed without causing a reactor trip.

Required Action B.1 provides for de-energizing the affected clutch power supplies associated with the inoperable channel within a Completion Time of 1 hour.

BASES

ACTIONS (continued)

C.1

Condition C applies to the failure of one Manual Trip channel. With one manual reactor trip channel inoperable operation may continue until the reactor is shut down for other reasons. Repair during operation is not required because one OPERABLE channel is all that is required for safe operation. No safety analyses assume operation of the Manual trip.

The Manual Trip channels are not testable without actually causing a reactor trip, so even if the difficulty were corrected, the post maintenance testing necessary to declare the channel OPERABLE could not be completed during operation. Because of this, the Required Action is to restore the inoperable channel to OPERABLE status prior to entering MODE 2 from MODE 3 during the next plant startup.

D.1

Condition D applies to the failure of both Trip Initiation Logic channels affecting the same trip leg. The affected control rod drive clutch power supplies must be de-energized immediately. With both channels inoperable, the RPS Function is lost if the affected clutch power supplies are not de-energized. Therefore, immediate action is required to de-energize the affected clutch power supplies. The immediate Completion Time is appropriate since there could be a loss of safety function if the associated clutch power supplies are not de-energized.

E.1, E.2.1 and E.2.2

Condition E is entered if Required Actions associated with Condition A, B, C, or D are not met within the required Completion Time or if for one or more Functions more than one Manual Trip, Matrix Logic, or Trip Initiation Logic channel is inoperable for reasons other than Condition D.

In Condition E the reactor must be placed in a MODE in which the LCO does not apply. The Completion Time of 6 hours to be in MODE 3 is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

E.1, E.2.1 and E.2.2 (continued)

Required Actions E.2.1 and E.2.2 allow 6 hours to verify that no more than one full-length control rod is capable of being withdrawn or to verify that PCS boron concentration is at REFUELING BORON CONCENTRATION. The Completion Time is reasonable to place the plant in an operating condition in which the LCO does not apply.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

A CHANNEL FUNCTIONAL TEST on each RPS Logic channel 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 SR addresses the two tests associated with the RPS Logic: Matrix Logic and Trip Initiation Logic.

Matrix Logic Tests

These tests are performed one matrix at a time. They verify that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. The Matrix Logic tests will detect any short circuits around the bistable contacts in the coincidence logic such as may be caused by faulty bistable relay or trip channel bypass contacts.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1 (continued)

Trip Initiation Logic Tests

These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, de-energizing the affected set of clutch power supplies.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.2.2

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

The Manual Trip Function is not tested at power. However, the simplicity of this circuitry and the absence of drift concern makes this Frequency adequate. Additionally, operating experience has shown that these components usually pass the Surveillance when performed once within 7 days prior to each reactor startup.

REFERENCES

1. 10 CFR 50, Appendix A
2. 10 CFR 100
3. FSAR, Figure 7-1
4. FSAR, Section 7.2
5. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989
6. 10 CFR 50.67

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. Re-positions CV-3001 and CV-3002 to a predetermined throttled position.

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

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 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.2

A CHANNEL FUNCTIONAL TEST is performed 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.

This test is required to be performed 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 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 3.3 INSTRUMENTATION

B 3.3.4 Engineered Safety Features (ESF) Logic and Manual Initiation

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 following 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 Signal (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

BASES

BACKGROUND (continued)

5. Containment High Pressure Signal (CHP);

   a. Containment High Pressure - Left Train

   b. Containment High Pressure - Right Train

6. Containment High Radiation Signal (CHR)

   a. Containment High Radiation

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 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. The actuation relays are considered part of the actuation logic addressed by this LCO.

RAS logic consists of output contacts of the relays actuated by the SIRWT Low 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 sensor subsystem, including individual channel bistables, is addressed in LCO 3.3.3, “Engineered Safety Features (ESF) Instrumentation.” This LCO addresses the actuation subsystem manual actuation, and downstream components used to actuate the individual ESF functions, as defined in the following section.

BASES

BACKGROUND (continued)

ESF Logic

Each of the six ESF actuation signals in Table 3.3.4-1 operates two trains of actuating relays. Each train is capable of initiating the ESF equipment to meet the minimum requirements to provide all functions necessary to operate the system associated with the plant's capability to cope with abnormal events.

The SGLP logic circuitry includes provisions such that the SGLP automatic actuation Function may be bypassed if three-out-of-four Steam Generator (SG) pressure channels are below a bypass permissive setpoint. Similarly, the SIS automatic actuation on Pressurizer Low Pressure may be bypassed when three-out-of-four channels are below a permissive setpoint. This actuation bypassing is performed when the ESF Functions are no longer required for protection. These actuation bypasses are enabled manually when the permissive conditions are satisfied.

All actuation bypasses are automatically removed when enabling conditions are no longer satisfied. If an SIS or SGLP automatic actuation channel is bypassed, other than as allowed by Table 3.3.4-1, the channel cannot perform its required safety function and must be considered to be inoperable.

Testing of a major portion of the ESF circuits is accomplished while the plant is at power. More extensive sequencer and load testing may be done with the reactor shut down. The test circuits are designed to test the redundant circuits separately such that the correct operation of each circuit may be verified by either equipment operation or by sequence lights.

Manual Initiation

Manual ESF initiation capability is provided to permit the operator to manually actuate an ESF System when necessary.

Two control room mounted manual actuation switches are provided for SIS actuation, one for each train. Each SIS manual actuation switch affects one actuation channel, which actuates one train of SIS equipment.

There are no single manual controls provided to actuate CHP, however, CHP may be manually initiated using individual component controls.

BASES

BACKGROUND (continued)

Manual Initiation (continued)

Two control room mounted manual actuation switches are provided for CHR actuation, each switch affects both actuation channels, which actuates both CHR trains.

There are no single manual controls provided to actuate SGLP, however, SGLP may be manually initiated using individual component controls.

RAS is actuated by manually actuating the circuit "Test" switch, however, RAS may also be manually initiated using individual component controls.

Manual actuation of AFW may be accomplished through pushbutton actuation of each AFAS channel or by use of individual pump and valve controls. Each automatic AFAS actuation channel starts the AFW pumps in their starting sequence (if P-8A fails to start, a P-8C start signal is generated, and if P-8C also fails to start, a P-8B start signal is generated) and opens the associated flow control valves.

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 such as Manual Initiation, not specifically credited in the accident analysis, serve as backups to Functions and are part of the NRC staff approved licensing basis for the plant.

The manual initiation is not required by the accident analysis. The ESF logic must function in all situations where the ESF function is required (as discussed in the Bases for LCO 3.3.3).

Each ESF Function and its associated safety analyses are discussed in the Applicable Safety Analyses section of the Bases for LCO 3.3.3, ESF Instrumentation.

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

BASES

LCO

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

The Bases for the LCO on ESF automatic actuation Functions are addressed in LCO 3.3.3. Those associated with the Manual Initiation or Actuation Logic are addressed below.

ESF Logic and Manual Initiation Functions are required to be OPERABLE in MODES 1, 2, and 3, or in MODES 1, 2, 3, and 4, as appropriate, when the associated automatic initiation channels addressed by LCO 3.3.3 are required.

1. Safety Injection Signal (SIS)

SIS is actuated by manual initiation, by a CHP signal, or by two-out-of-four Pressurizer Low Pressure channels decreasing below the setpoint. Each Manual Initiation channel consists of one pushbutton which directly starts the SIS actuation logic for the associated train. Each SIS logic train is also actuated by a contact pair on one of the CHP initiation relays for the associated CHP train.

a. Manual Initiation

This LCO requires two channels of SIS Manual Initiation to be OPERABLE.

b. Actuation Logic

This LCO requires two channels of SIS Actuation Logic to be OPERABLE. Failures in the actuation subsystems are addressed in this LCO.

c. CHP Logic Trains

The CHP initiation relay (5P-x) input to the SIS logic is considered part of the SIS logic. Two channels, one per SIS train, must be OPERABLE.

BASES

LCO (continued)

1. Safety Injection Signal (SIS) (continued)

d. Automatic Bypass Removal

This LCO requires two channels of the automatic bypass removal logic for SIS Pressurizer Low Pressure to be OPERABLE. If an SIS automatic actuation channel is bypassed, other than as allowed by Table 3.3.4-1, the channel cannot perform its required safety function and must be considered to be inoperable.

As indicated by footnote (a), the Pressurizer Low Pressure logic train for each SIS train can be bypassed when three-out-of-four channels indicate below 1700 psia. This bypass prevents undesired actuation of SIS during a normal plant cooldown. The bypass signal is automatically removed when two-out-of-four channels exceed the setpoint, in accordance with the philosophy of removing bypasses when the enabling conditions are no longer satisfied.

The bypass permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow bypassing prior to reaching the trip setpoint.

2. Steam Generator Low Pressure Signal (SGLP)

a. Manual Initiation

This LCO requires two channels of SGLP Manual Initiation to be OPERABLE. As indicated by footnote (c), there is no manual control which actuates the SGLP logic circuits. The actuated components must be individually actuated using control room manual controls.

b. Actuation Logic

This LCO requires two channels of SGLP Actuation Logic to be OPERABLE, one for each SG.

BASES

LCO (continued)

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

c. Automatic Bypass Removal

This LCO requires two channels, one for each SG, of the SGLP automatic bypass removal logic to be OPERABLE. If an SGLP automatic actuation channel is bypassed, other than as allowed by Table 3.3.4-1, the channel cannot perform its required safety function and must be considered to be inoperable.

As indicated by footnote (b), the SGLP from each SG may be bypassed when three-out-of-four channels indicate below 565 psia. This bypass prevents undesired actuation during a normal plant cooldown. The bypass signal is automatically removed when two-out-of-four channels exceed the setpoint, in accordance with the philosophy of removing bypasses when the enabling conditions are no longer satisfied.

The bypass permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow bypassing prior to reaching the trip setpoint.

3. Recirculation Actuation Signal (RAS)

a. Manual Initiation

This LCO requires two channels of RAS Manual Initiation to be OPERABLE. RAS is actuated by manually actuating the circuit "Test" switches.

b. Actuation Logic

This LCO requires two channels of RAS Actuation Logic to be OPERABLE.

BASES

LCO (continued)

4. Auxiliary Feedwater Actuation Signal (AFAS)

a. Manual Initiation

This LCO requires two channels of AFAS Manual Initiation to be OPERABLE. Each train of AFAS may be manually initiated with either of two sets of controls. Only one set of manual controls is required to be OPERABLE for each AFW train. One set of controls are the pushbuttons provided to actuate each train on the C-11 panel; the other set of controls are those manual controls provided on C-01 for each AFW pump and flow control valve.

b. Actuation Logic

This LCO requires two channels of AFAS Actuation Logic to be OPERABLE.

5. Containment High Pressure Signal (CHP)

a. Manual Initiation

As indicated by footnote (c), this LCO requires the manual controls necessary to actuate those valves and components actuated by an automatic CHP to be OPERABLE.

b. Actuation Logic

This LCO requires two channels of CHP Actuation Logic to be OPERABLE.

6. Containment High Radiation Signal (CHR)

a. Manual Initiation

This LCO requires two channels of CHR Manual Initiation to be OPERABLE. Pushbuttons are available for manual actuation of each CHR logic train.

BASES

LCO (continued)

6. Containment High Radiation Signal (CHR) (continued)

b. Actuation Logic

This LCO requires two channels of CHR Actuation Logic to be OPERABLE.

APPLICABILITY

ESF Functions are required to be OPERABLE in MODES 1, 2, and 3 or MODES 1, 2, 3, and 4 as specified in Table 3.3.4-1. 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 MSIVs 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 also 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 function under these conditions.

BASES

APPLICABILITY (continued)

In MODES 5 and 6, 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 if required. In these MODES, ESF 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

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 the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.4-1 in the LCO. Completion Times for the inoperable channel of a Function will be tracked separately.

A.1

Condition A applies to one Manual Initiation, Bypass Removal, or Actuation Logic channel inoperable. The channel must be restored to OPERABLE status to restore redundancy of the ESF Function. The 48 hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel.

B.1 and B.2

If two Manual Initiation, Bypass Removal, or Actuation Logic channels are inoperable for Functions 1, 2, 3, or 4, or if the Required Action and associated Completion Time of Condition A cannot be met for Function 1, 2, 3, or 4, the reactor must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 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.

BASES

ACTIONS (continued)

C.1 and C.2

Condition C is entered when one or more Functions have two Manual Initiation or Actuation Logic channels inoperable for Functions 5 or 6, or when the Required Action and associated Completion Time of Condition A are not met for Functions 5 or 6. If Required Action A.1 cannot be met within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

A SIS actuation functional test of each channel is performed using the installed control room test switches and test circuits for both “with standby power” and “without standby power”. When testing the “with standby power” circuits, proper operation of the “SIS-X” relays must be verified; when testing the “without standby power” circuits, proper operation of the “DBA sequencer” and the associated logic circuit must be verified. The test circuits are designed to block those SIS functions, such as injection of concentrated boric acid, which would interfere with plant operation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.2

A CHANNEL FUNCTIONAL TEST of each AFAS Actuation Logic Channel is performed to ensure the 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.2 (continued)

Instrumentation channel tests are addressed in LCO 3.3.3.

SR 3.3.4.2 addresses Actuation Logic tests of the AFAS using the installed test circuits.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.3

A CHANNEL FUNCTIONAL TEST is performed on the manual ESF initiation channels, Actuation Logic channels, and bypass removal channels for specified ESF Functions. 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 Surveillance verifies that the required channels will perform their intended functions when needed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 7

2. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989

B 3.3 INSTRUMENTATION

B 3.3.5 Diesel Generator (DG) - Undervoltage Start (UV Start)

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or insufficiently stable to allow safe plant operation. Undervoltage protection will generate a UV Start in the event a Loss of Voltage or Degraded Voltage condition occurs. There are two UV Start Functions for each 2.4 kV vital bus.

Undervoltage protection and load shedding features for safety-related buses at the 2,400 V and lower voltage levels are designed in accordance with 10 CFR 50, Appendix A, General Design Criterion 17 (Ref. 1) and the following features:

1. Two levels of automatic undervoltage protection from loss or degradation of offsite power sources are provided. The first level (loss of voltage) provides normal loss of voltage protection. The second level of protection (degraded voltage) has voltage and time delay set points selected for automatic trip of the offsite sources to protect safety-related equipment from sustained degraded voltage conditions at all bus voltage levels. Coincidence logic is provided to preclude spurious trips.

2. The undervoltage protection system automatically prevents load shedding of the safety-related buses when the emergency generators are supplying power to the safeguards loads.

3. Control circuits for shedding of Class 1E and non-Class 1E loads during a Loss of Coolant Accident (LOCA) themselves are Class 1E or are separated electrically from the Class 1E portions.

BASES

BACKGROUND (continued)

Description

Each 2,400 V Bus (1C and 1D) is equipped with two levels of undervoltage protection relays (Ref. 2). The first level (Loss of Voltage Function) relays 127-1 and 127-2 are set at approximately 77% of rated voltage with an inverse time relay. One of these relays measures voltage on each of the three phases. They protect against sudden loss of voltage as sensed on the corresponding bus using a three-out-of-three coincidence logic. The actuation of the associated auxiliary relays will trip the associated bus incoming circuit breakers, start its associated DG, initiate bus load shedding, and activate annunciators in the control room. The DG circuit breaker is closed automatically upon establishment of satisfactory voltage and frequency by the use of associated voltage sensing relay 127D-1 or 127D-2.

The second level of undervoltage protection (Degraded Voltage Function) relays 127-7 and 127-8 are set at approximately 92% of rated voltage, with one relay monitoring each of the three phases. These voltage sensing relays protect against sustained degraded voltage conditions on the corresponding bus using a three-out-of-three coincidence logic. These relays have an internal (built-in) 0.65 second time delay, after which the associated DG receives a start signal and annunciators in the control room are actuated. If the bus undervoltage condition exists for an additional six seconds (due to a six-second time delay relay), the associated bus incoming circuit breakers will be tripped and a bus load shed will be initiated.

Trip Setpoints

The trip setpoints are based on the analytical limits discussed in References 3, 4, 5, 7, 9, and 10. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, setpoints specified in SR 3.3.5.2 are conservatively adjusted with respect to the analytical limits. A detailed analysis of the degraded voltage protection is provided in References 3 and 4.

The specified setpoints will ensure that the consequences of accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the accident and the equipment functions as designed.

BASES

APPLICABLE SAFETY ANALYSES

The DG - UV Start is required for Engineered Safety Features (ESF) systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Systems.

Accident analyses credit the loading of the DG based on a loss of offsite power during a LOCA. The diesel loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. This delay time includes contributions from the DG start, DG loading, and Safety Injection System component actuation.

The required channels of UV Start, in conjunction with the ESF systems powered from the DGs, provide plant protection in the event of any of the analyzed accidents discussed in Reference 6, in which a loss of offsite power is assumed. UV Start channels are required to meet the redundancy and testability requirements of GDC 21 in 10 CFR 50, Appendix A (Ref. 1).

The delay times assumed in the safety analysis for the ESF equipment include the 10 second DG start delay and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment include the appropriate DG loading and sequencing delay.

The DG - UV Start channels satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The LCO for the DG - UV Start requires that three channels per bus of each UV Start instrumentation Function be OPERABLE when the associated DG is required to be OPERABLE. The UV Start supports safety systems associated with ESF actuation.

The Bases for the trip setpoints are as follows:

The voltage trip setpoint is set low enough such that spurious trips of the offsite source due to operation of the undervoltage relays are not expected for any combination of plant loads and normal grid voltages.

BASES

LCO (continued)

This setpoint at the 2,400 V bus and reflected down to the 480 V buses has been verified through an analysis to be greater than the minimum allowable motor voltage (90% of nominal voltage). Motors are the most limiting equipment in the system. MCC contactor pickup and drop-out voltage is also adequate at the setpoint values. The analysis ensures that the distribution system is capable of starting and operating all safety-related equipment within the equipment voltage rating at the allowed source voltages. The power distribution system model used in the analysis has been verified by actual testing (Refs. 5 and 7).

The time delays involved will not cause any thermal damage as the setpoints are within voltage ranges for sustained operation. They are long enough to preclude trip of the offsite source caused by the starting of large motors and yet do not exceed the time limits of ESF actuation assumed in FSAR Chapter 14 (Ref. 6) and validated by Reference 8. The time delays also will not result in failure of safety related equipment due to sustained degraded voltage conditions (Reference 9).

Calibration of the undervoltage relays verify that the time delays are sufficient to avoid spurious trips.

APPLICABILITY

The DG - UV Start actuation Function is required to be OPERABLE whenever the associated DG is required to be OPERABLE per LCO 3.8.1, “AC Sources - Operating,” or LCO 3.8.2, “AC Sources - Shutdown,” so that it can perform its function on a loss of power or degraded power to the vital bus.

ACTIONS

A DG - UV Start channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function.

In the event a channel's trip setpoint is found nonconservative with respect to the specified setpoint, or the channel is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition entered. The required channels are specified on a per DG basis.

BASES

ACTIONS (continued)

A.1

Condition A applies if one or more of the three phase UV sensors or relay logic is inoperable for one or more Functions (Degraded Voltage or Loss of Voltage) per DG bus.

The affected DG must be declared inoperable and the appropriate Condition(s) entered. Because of the three-out-of-three logic in both the Loss of Voltage and Degraded Voltage Functions, the appropriate means of addressing channel failure is declaring the DG inoperable, and effecting repair in a manner consistent with other DG failures.

Required Action A.1 ensures that Required Actions for the affected DG inoperabilities are initiated. Depending upon plant MODE, the actions specified in LCO 3.8.1 or LCO 3.8.2, as applicable, are required immediately.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

A CHANNEL FUNCTIONAL TEST is performed on each UV Start logic channel to ensure that the logic channel will perform its intended function when needed. The Undervoltage sensing relays are tested by SR 3.3.5.2. 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.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.2

A CHANNEL CALIBRATION verifies the accuracy of each component within the instrument channel. This includes calibration of the undervoltage relays and demonstrates that the equipment falls within the specified operating characteristics defined by the manufacturer.

The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Degraded Voltage Function time delay setpoints reflect the voltage sensing relay nominal 0.65-second time delay, and the voltage sensing relay nominal 0.65-second time delay combined with the nominal six-second delay due to the external time delay relay.

CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the setpoint analysis.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A GDCs 17 and 21
2. FSAR, Section 8.6
3. Analysis EA-ELEC-VOLT-033
4. Analysis EA-ELEC-VOLT-034
5. Analysis EA-ELEC-EDSA-04
6. FSAR, Chapter 14
7. Analysis EA-ELEC-EDSA-03
8. Analysis A-NL-92-111
9. Analysis 0098-0189-CALC-001-PLP
10. Analysis EA-EC11464-01

B 3.3 INSTRUMENTATION

B 3.3.6 Refueling Containment High Radiation (CHR) Instrumentation

BASES

BACKGROUND

This LCO addresses Refueling CHR actuation. When the Refueling CHR Monitors are enabled by their keylock switches, a CHR actuation may be automatically initiated by a signal from either of the Refueling CHR monitors or manually by actuation of either of the control room “CHR Manual Initiate” pushbuttons (pushing either Manual Initiate pushbutton will actuate both trains of CHR). A CHR signal initiates the following actions:

a. Control Room HVAC Emergency Mode;

b. Containment Isolation Valve Closure; and

c. Block automatic starting of Engineered Safeguards pump room sump pumps.

The Refueling CHR signal provides automatic containment isolation valve closure during refueling operations, using two radiation monitors located in the refueling area of the containment (elevation 649 ft). The monitors are part of the plant area monitoring system and employ one-out-of-two logic for isolation. During normal operation these monitors are disconnected from the CHR relays and will not initiate a CHR signal. A switch is provided to connect the Refueling CHR monitors into the CHR actuation circuit, so that CHR actuation can be initiated by these monitors during refueling.

BASES

BACKGROUND (continued)

Each monitor actuates one train of CHR logic when containment radiation exceeds the setpoint. Two separate keylock switches, one per train, enable the Refueling CHR input to the CHR logic when switched to the "Refueling" position. Each Refueling CHR channel, associated keylock switch, and initiation circuit input to the CHR logic thus forms a one-out-of-one logic input to its associated CHR actuation logic train. The Refueling CHR isolation instrumentation is separate from the CHR instrumentation addressed in LCO 3.3.3, "ESF Instrumentation." However, the Refueling CHR Instrumentation does operate the same CHR actuation relays as the two-out-of-four CHR logic addressed in LCO 3.3.4. This LCO is not included in LCOs 3.3.3 and 3.3.4 because of the differences in APPLICABILITY and the single channel nature of the Refueling CHR input. The Refueling CHR signal performs the automatic containment isolation valve closure Function during refueling operations required by LCO 3.9.3, "Containment Penetrations."

The Refueling CHR Instrumentation provides protection from release of radioactive gases and particulates from the containment in the event a fuel assembly should be severely damaged during handling.

The Refueling CHR Instrumentation will detect any abnormal radiation levels in the containment refueling area and will initiate purge valve closure to limit the release of radioactivity to the environment. The containment purge supply and exhaust valves are closed on a CHR signal when a high radiation level in containment is detected.

The Refueling CHR Instrumentation includes two independent, redundant actuation subsystems, as described above. Reference 1 describes the Refueling CHR circuitry.

Trip Setpoint

No required setpoint is specified because these instruments are not assumed to function by any of the safety analyses. Typically, the instruments are set at about 25 mR/hr above expected background for planned operations (including movement of the reactor vessel head or internals).

BASES

APPLICABLE SAFETY ANALYSES	The Refueling CHR Instrumentation isolates containment in the event that area radiation exceeds an established level following a fuel handling accident. This ensures the radioactive materials are not released directly to the environment and significantly reduces the offsite doses from those calculated by the safety analyses, which do not credit containment isolation (Ref. 2). Either way, i.e., with or without containment isolation, the offsite doses remain within applicable 10 CFR 50.67 limits. The Refueling CHR Instrumentation is not required by the fuel handling accident analyses to maintain offsite doses within applicable 10 CFR 50.67 limits, but containment isolation would provide a significant reduction of the resulting offsite doses. Therefore, the Refueling CHR Instrumentation satisfies the requirements of Criterion 4 of 10 CFR 50.36(c)(2).
LCO	The LCO for the Refueling CHR Instrumentation requires that two channels of refueling CHR instrumentation and two channels of CHR manual initiation be OPERABLE, including the logic components necessary to initiate Refueling CHR Isolation. The CHR setpoint is chosen to be high enough to avoid inadvertent actuation in the event of normal background radiation fluctuations during fuel handling and movement of the reactor internals, but low enough to alarm and isolate the containment in the event of a Design Basis fuel handling accident.
APPLICABILITY	In MODE 5 or 6, the Refueling CHR isolation of containment isolation valves is not normally required to be OPERABLE. However, during CORE ALTERATIONS or during movement of irradiated fuel within containment, there is the possibility of a fuel handling accident requiring containment isolation on high radiation in containment. Accordingly, the Refueling CHR Instrumentation must be OPERABLE during CORE ALTERATIONS and when moving any irradiated fuel in containment. In MODES 1, 2, 3 and 4, both the Containment High Pressure (CHP) and CHR signals provide containment isolation as discussed in the Bases for LCO 3.3.3 and LCO 3.3.4.

BASES

ACTIONS

A.1, A.2.1, and A.2.2

Condition A applies to the failure of one Refueling CHR monitor channel, one CHR Manual Initiate channel, or one of each. The Required Action allows either initiation of a CHR signal by placing the inoperable channel in trip (which accomplishes the safety function of the inoperable channel), or suspension of CORE ALTERATIONS and movement of irradiated fuel assemblies within containment (which places the plant in a condition where the LCO does not apply). The Completion Time of 4 hours is acceptable because one additional channel of each Function remains operable during that period and the probability of an additional failure occurring during this period is very small.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

B.1 and B.2

Condition B applies when either no automatic Refueling CHR or no Manual CHR (or neither) is available. The Required Action is to immediately suspend CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. This places the plant in a condition where the LCO does not apply. The Completion Time is warranted on the basis that at least one containment isolation Function is completely lost.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

Performance of the CHANNEL CHECK 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1 (continued)

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or actual differing radiation levels at the two detector locations. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.2

A CHANNEL FUNCTIONAL TEST is performed on each Refueling CHR channel to ensure the entire channel will perform its intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.3

A CHANNEL FUNCTIONAL TEST is performed on each CHR Manual Initiation channel to ensure it will perform its intended function.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.4

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

No required setpoint is specified because these instruments are not assumed to function by any of the safety analyses.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 7.3

2. FSAR, Section 14.19

B 3.3 INSTRUMENTATION

B 3.3.7 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the Post Accident Monitoring (PAM) instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety Functions for Design Basis Events.

The OPERABILITY of the PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident.

The availability of PAM instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. The required instruments are identified in FSAR Appendix 7C (Ref. 1) and address the recommendations of Regulatory Guide 1.97 (Ref. 2), as required by Supplement 1 to NUREG-0737, "TMI Action Items" (Ref. 3).

Type A variables are included in this LCO because they provide the primary information required to permit the control room operator to take specific manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

Category I variables are the key variables deemed risk significant because they are needed to:

- Determine whether other systems important to safety are performing their intended functions;
- Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

BASES

BACKGROUND (continued)

These key variables are identified in the plant specific Regulatory Guide 1.97 analyses (Ref. 1). This analysis identified the plant specific Type A and Category 1 variables and provided justification for deviating from the NRC proposed list of Category I variables.

The specific instrument Functions listed in Table 3.3.7-1 are discussed in the LCO Bases.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the OPERABILITY of Regulatory Guide 1.97 Type A variables, so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of DBAs; and
- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions.

The PAM instrumentation also ensures OPERABILITY of Category I, non-Type A variables. This ensures the control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;
- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public as well as to obtain an estimate of the magnitude of any impending threat.

Category I, non-Type A PAM instruments are retained in the Specification because they are intended to assist operators in minimizing the consequences of accidents. Therefore, these Category I variables are important in reducing public risk.

PAM instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 meets Criterion 3 of 10 CFR 50.36(c)(2).

BASES

LCO

LCO 3.3.7 requires at least two OPERABLE channels for all Functions except Containment Isolation Valve Position Indication. This is to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following that accident.

Furthermore, provision of at least two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

For Containment Isolation Valve Position indication, the important information is the status of the containment penetrations. The LCO requires one position indication channel for each containment isolation valve listed in FSAR Appendix 7C (Ref. 1).

Listed below are discussions of the specified instrument Functions listed in Table 3.3.7-1. Component identifiers of the sensors, indicators, power supplies, displays, and recorders in each instrument loop are found in Reference 1.

1, 2. Primary Coolant System (PCS) Hot and Cold Leg Temperature (wide range)

PCS wide range Hot and Cold Leg Temperatures are Type B, Category 1 variables provided for verification of core cooling and long term surveillance.

Reactor outlet temperature inputs to the PAM are provided by two wide range resistance elements and associated transmitters (one in each loop). The channels provide indication over a range of 50°F to 700°F.

3. Wide Range Neutron Flux

Wide Range Neutron Flux indication is a Type B, Category 1 variable, and is provided to verify reactor shutdown.

4. Containment Floor Water Level (wide range)

Wide range Containment Floor Water Level is a Type B, Category 1 variable, and is provided for verification and long-term surveillance of PCS integrity.

BASES

LCO (continued)

5. Subcooled Margin Monitor

The Subcooled Margin Monitor (SMM) is a Type A, Category 1 variable used to identify conditions, which require tripping of the primary coolant pumps and throttling of safety injection flows. Each SMM channel uses a number of PCS pressure and temperature inputs to determine the degree of PCS subcooling or superheat.

6. Pressurizer Level (Wide Range)

Pressurizer Level is a Type A, Category 1 variable, and is used to determine whether to terminate Safety Injection (SI), if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the plant conditions necessary to establish natural circulation in the PCS and to verify that the plant is maintained in a safe shutdown condition.

7. (Deleted)

8. Condensate Storage Tank (CST) Level

CST Level is a Type D, Category 1 variable, and is provided to ensure water supply for AFW. The CST provides the safety grade water supply for the AFW System. Inventory is monitored by a 0 to 100% level indication. CST Level is displayed on a control room indicator. In addition, a control room annunciator alarms on low level.

The CST is the initial source of water for the AFW System. However, as the CST is depleted, manual operator action is necessary to replenish the CST.

BASES

LCO (continued)

9. Primary Coolant System Pressure (wide range)

PCS wide range pressure is a Type A, Category 1 variable provided for verification of core cooling and PCS integrity long-term surveillance.

Wide range PCS loop pressure is measured by pressure transmitters with a span of 0 psia to 3000 psig. Redundant monitoring capability is provided by two channels of instrumentation. Control room indications are provided on C12 and C02.

10. Containment Pressure (wide range)

Wide range Containment Pressure is a Type C, Category 1 variable, and is provided for verification of PCS and containment OPERABILITY. It is also an input to decisions for initiating containment spray.

11, 12. Steam Generator Water Level (wide range)

Wide range Steam Generator Water Level is a Type A, Category 1 variable, and is provided to monitor operation of decay heat removal via the steam generators. The steam generator level instrumentation covers a span extending from the tube sheet to the steam separators, with an indicated range of -140% to +150%. Redundant monitoring capability is provided by two channels of instrumentation for each SG.

Operator action for maintenance of heat removal is based on the control room indication of Steam Generator Water Level. The indication is used during a SG tube rupture to determine which SG has the ruptured tube. It is also used to determine when to initiate once through cooling on low water level.

13, 14. SG Pressure

Steam Generator Pressure is a Type A, Category 1 variable used in accident identification, including Loss of Coolant, and Steam Line Break. Redundant monitoring capability is provided by two channels of instrumentation for each SG.

BASES

LCO (continued)

15. Containment Isolation Valve Position

Containment Isolation Valve (CIV) Position is a Type B, Category 1 variable, and is provided for verification of containment OPERABILITY.

CIV position is provided for verification of containment integrity. In the case of CIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each CIV listed in FSAR Appendix 7C (Ref. 1). This is sufficient to redundantly verify the isolation status of each associated penetration via indicated status of the CIVs, and by knowledge of a passive (check) valve or a closed system boundary.

If a penetration flow path is isolated, position indication for the CIV(s) in the associated penetration flow path is not needed to determine status. Therefore, as indicated in Note (a) the position indication for valves in an isolated penetration flow path is not required to be OPERABLE.

16, 17, 18, 19. Core Exit Temperature

Core Exit Temperature is a Type C, Category 1 variable, and is provided for verification and long term surveillance of core cooling.

Each Required Core Exit Thermocouple (CET) channel consists of a single environmentally qualified thermocouple.

The design of the Incore Instrumentation System includes a Type K (chromel alumel) thermocouple within each of the incore instrument detector assemblies.

The junction of each thermocouple is located above the core exit, inside the incore detector assembly guide tube, that supports and shields the incore instrument detector assembly string from flow forces in the outlet plenum region. These core exit thermocouples monitor the temperature of the reactor coolant as it exits the fuel assemblies.

The core exit thermocouples have a usable temperature range from 32°F to 2300°F, although accuracy is reduced at temperatures above 1800°F.

BASES

LCO (continued)

20. Reactor Vessel Water Level

Reactor Vessel Water Level is monitored by the Reactor Vessel Level Monitoring System (RVLMS) and is a Type B, Category 1 variable provided for verification and long-term surveillance of core cooling.

The RVLMS provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory. The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it functions during the recovery interval. Therefore, it is designed to survive the high steam temperature that may occur during the preceding core recovery interval.

The level range extends from the top of the vessel down to the top of the fuel alignment plate. A total of eight Heated Junction Thermocouple (HJTC) pairs are employed in each of the two RVLMS channels. Each pair consists of a heated junction TC and an unheated junction TC. The differential temperature at each HJTC pair provides discrete indication of uncovery at the HJTC pair location. This indication is displayed using LEDs in the control room. This provides the operator with adequate indication to track the progression of the accident and to detect the consequences of its mitigating actions or the functionality of automatic equipment.

A RVLMS channel consists of eight sensors in a probe. A channel is OPERABLE if four or more sensors, two or more of the upper four and two or more of the lower four, are OPERABLE.

21. Containment Area Radiation (high range)

High range Containment Area Radiation is a Type E, Category 1 variable, and is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

BASES

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, plant conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

A note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.7-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

A.1

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

B.1

This Required Action specifies initiation of actions in accordance with Specification 5.6.6, which requires a written report to be submitted to the Nuclear Regulatory Commission. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative Required Actions. This Required Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Required Actions are identified before a loss of functional capability condition occurs.

C.1

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of

ACTIONS (continued)

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

D.1

Condition D is currently not used.

E.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.7-1. The applicable Condition referenced in the Table is Function dependent. Each time Required Action C.1 is not met, and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1 and F.2

If the Required Action and associated Completion Time of Condition C is not met, and Table 3.3.7-1 directs entry into Condition F, the plant must be brought to a MODE in which the requirements of this LCO do 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.

G.1

Alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the plant, but rather to follow the directions of Specification 5.6.6. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

BASES

SURVEILLANCE REQUIREMENTS

A Note at the beginning of the Surveillance Requirements specifies that the following SRs apply to each PAM instrumentation Function in Table 3.3.7-1.

SR 3.3.7.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verify 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. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.2

A CHANNEL CALIBRATION is typically a complete check of the instrument channel including the sensor. Therefore, this SR is modified by a Note, which states that it is not necessary to calibrate neutron detectors because of the difficulty of simulating a meaningful signal. Wide range and source range nuclear instrument channels are not calibrated to indicate the actual power level or the flux in the detector location. The circuitry is adjusted so that wide range and source range readings may be used to determine the approximate reactor flux level for comparative purposes. The Surveillance verifies the channel responds to the measured parameter within the necessary range and accuracy.

For the core exit thermocouples, a CHANNEL CALIBRATION is performed by substituting a known voltage for the thermocouple.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Appendix 7C, “Regulatory Guide 1.97 Instrumentation”

2. Regulatory Guide 1.97

3. NUREG-0737, Supplement 1

B 3.3 INSTRUMENTATION

B 3.3.8 Alternate Shutdown System

BASES

BACKGROUND

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

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

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

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

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

1. AFW flow control valves (CV-0727 and CV-0749); and

2. Turbine-driven AFW pump.

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

1. Source range flux monitor;

2. AFW flow (HIC-0727 and HIC-0749C);

3. Pressurizer pressure;

4. Pressurizer level;

5. SG level and pressure;

6. Primary coolant temperatures (hot and cold legs);

7. Turbine-driven AFW pump low-suction pressure warning light; and

8. SIRW tank level.

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

BASES

LCO (continued)

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

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

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

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

APPLICABILITY

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

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

BASES

ACTIONS

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

A.1

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

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

B.1 and B.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

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

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

SR 3.3.8.2

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.3

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

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

REFERENCES

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

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

B 3.3 INSTRUMENTATION

B 3.3.9 Neutron Flux Monitoring Channels

BASES

BACKGROUND

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

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

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

APPLICABLE SAFETY ANALYSES

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

BASES

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

INSERT Bases 3.3.9

BASES

ACTIONS

A.1 and A.2

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

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

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

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

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

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.2

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

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

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13
2. FSAR, Chapter 7
3. FSAR, Chapter 14

B 3.3 INSTRUMENTATION

B 3.3.10 Engineered Safeguards Room Ventilation (ESRV) Instrumentation

BASES

BACKGROUND	This LCO addresses the instrumentation which provides isolation of the ESRV System (Ref. 1). The ESRV Instrumentation high radiation signal provides automatic damper closure, using two radiation monitors. One radiation monitor is located in the ventilation system duct work associated with each of the Engineered Safeguards (ES) pump rooms. Upon detection of high radiation, the ESRV Instrumentation actuates isolation of the associated ES pump room by closing the dampers in the ventilation system inlet and discharge paths. Typically, high radiation would only be expected due to excessive leakage during the recirculation phase of operation following a Loss of Coolant Accident (LOCA). The ESRV System is addressed by LCO 3.7.13, "Engineered Safeguards Room Ventilation (ESRV) Dampers."
APPLICABLE SAFETY ANALYSES	The ESRV Instrumentation isolates the ES pump rooms in the event of high radiation in the pump rooms due to leakage during the recirculation phase. The analysis for a Maximum Hypothetical Accident (MHA) described in FSAR, Section 14.22 (Ref. 2), assumes a reduction factor in the potential radioactive releases from the ES pump rooms due to plateout following automatic isolation. However, no specific value is assumed in the MHA for the timing of actuation of the isolation. The results indicate that the potential MHA offsite doses would be less than applicable 10 CFR 50.67 limits. The ESRV Instrumentation satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2).
LCO	The LCO for the ESRV Instrumentation requires both channels to be OPERABLE to initiate ES pump room isolation when high radiation exceeds the trip setpoint. The ESRV Instrumentation Setpoint is specified as ≤ 2.2E+5 cpm. This setpoint is high enough to avoid inadvertent actuation in the event of normal background radiation fluctuations during testing, but low enough to isolate the ES pump room in the event of radiation levels indicative of a LOCA and excessive leakage during recirculation of primary coolant through the ES pump room.

BASES

APPLICABILITY

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

While in MODE 5 and in MODE 6, the ESRV Instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the applicable 10 CFR 50.67 limits.

ACTIONS

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

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

A.1

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

INSERT Bases 3.3.10

BASES

ACTIONS (continued)

A.1 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

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

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

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.10.2

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

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.10.3

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 7.4.5.2

2. FSAR, Section 14.22

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

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

BASES

BACKGROUND

These Bases address requirements for maintaining PCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters, when appropriate measurement uncertainties are applied, will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum Departure from Nucleate Boiling Ratio (DNBR) will meet the required criteria for each of the transients analyzed.

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

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

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

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

BASES

APPLICABLE SAFETY ANALYSES	The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR Safety Limit (SL 2.1.1). This is the acceptance limit for the PCS DNB parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed for include loss of coolant flow events and dropped or stuck control rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Regulating Rod Group Position Limits"; LCO 3.2.3, "Quadrant Power Tilt"; and LCO 3.2.4, "AXIAL SHAPE INDEX." The PCS DNB limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).
LCO	This LCO specifies limits on the monitored process of variables PCS pressurizer pressure and PCS cold leg temperature, and the calculated value of PCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient. The LCO numerical values for pressure and temperature specified in the COLR are given for the measurement location but have not been adjusted for instrument error. Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of this LCO. Instrument errors and the PCS flow rate measurement error are applied to the LCO numerical values in the safety analysis.
APPLICABILITY	In MODE 1, the limits on PCS pressurizer pressure, PCS cold leg temperature, and PCS flow rate must be maintained during steady state operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough so that DNBR is not a concern.

BASES

ACTIONS

A.1

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

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

B.1

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 and SR 3.4.1.2

The Surveillance for monitoring pressurizer pressure and PCS cold leg temperature is performed using installed instrumentation. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.3

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

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.3 (continued)

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

REFERENCES

1. FSAR, Section 14.1

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.2 PCS Minimum Temperature for Criticality

BASES

BACKGROUND

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

a. Operation within the existing instrumentation ranges and accuracies;

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

LCO

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

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

BASES

APPLICABILITY	The reactor has been designed and analyzed to be critical in MODES 1 and 2 only and in accordance with this specification. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1, and MODE 2 when $K_{eff} \geq 1.0$.
ACTIONS	A.1 If T_{ave} is below 525°F and cannot be restored in 30 minutes, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with $K_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time reflects the ability to perform this action and to maintain the plant within the analyzed range.
SURVEILLANCE REQUIREMENTS	SR 3.4.2.1 PCS loop average temperature is required to be verified at or above 525°F. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. FSAR, Section 14.1.3

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

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

BASES

BACKGROUND

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

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

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

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

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

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

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the requirements of 10 CFR 50, Appendix G (Ref. 2) .

BASES

BACKGROUND (continued)

A discussion of the methodology for the development of the P/T limit curves is provided in Reference 1 and Reference 7. The P/T limit curves were originally developed to be valid up to an accumulated reactor vessel wall fluence at the limiting circumferential weld of 2.192 x 10^{19} n/cm^2 (E >1.0 MeV). It was subsequently determined that this fluence would be reached prior to the operating license expiration date. In order to continue to use the existing P/T limit curves, an evaluation (Ref. 8) using more recently approved NRC methods was performed to demonstrate that the P/T limit curves are valid through the operating license expiration date, equivalent to 42.1 Effective Full Power Years (EFPY). This evaluation was performed using the adjusted RT_{NDT} (ART) corresponding to the limiting beltline region material of the reactor vessel. The ART is defined as the sum of the initial reference temperature (RT_{NDT}) of the material, the mean value for the adjustment in RT_{NDT} caused by neutron irradiation, and a margin term to account for uncertainties in RT_{NDT}, percent nickel, percent copper, neutron fluence and calculational procedures (Ref. 9).

The specific input parameters below were used to validate that the existing P/T limit curves are conservative through an applicability period of 42.1 EFPY. The input parameters are for the limiting reactor vessel material, which are the intermediate and lower shell axial welds 2-112 and 3-112.

1. A peak reactor vessel wall surface fluence of 2.161 x 10^{19} n/cm^2 (E > 1.0 MeV)
2. ART values, at 1/4T = 252.7°F, and at 3/4T = 185.8°F
3. Initial RT_{NDT} = -56 °F
4. Margin term = 65.5 °F

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

BASES

BACKGROUND (continued)

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

The minimum temperature at which the reactor can be made critical, as required by Reference 2, shall be at least 40°F above the heatup curve or the cooldown curve and not less than the minimum permissible temperature for the ISLH testing. However, the criticality limit is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "PCS Minimum Temperature for Criticality," and LCO 3.1.7, "Special Test Exceptions (STE)."

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

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) Analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the PCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

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

BASES

LCO

The two elements of this LCO are:

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

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the PCS, except the pressurizer.

These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Additional cooldown rate restrictions were put in place due to the reactor vessel head nozzle repairs per Reference 7. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

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

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

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

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

APPLICABILITY

The PCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2) and due to the reactor vessel nozzle repairs (Ref. 7). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The additional cooldown rate restrictions for the reactor vessel nozzle repairs only apply when the reactor vessel head is on the reactor vessel. The limits do not apply to the pressurizer.

BASES

APPLICABILITY (continued)

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

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

ACTIONS

A.1 and A.2

Operation outside the P/T limits must be corrected so that the PCPB is returned to a condition that has been verified by stress analyses.

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

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

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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

BASES

ACTIONS (continued)

A.1 and A.2

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

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because:

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

b. A sufficiently severe event caused entry into an unacceptable region.

Either possibility indicates a need for more careful examination of the event, best accomplished with the PCS at reduced pressure and temperature. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is generally decreased.

Pressure and temperature are reduced by placing the plant in MODE 3 within 6 hours and in MODE 5 with PCS pressure < 270 psia within 36 hours.

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

BASES

ACTIONS (continued)

C.1 and C.2

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

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

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

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the limits of Figure 3.4.3-1 and Figure 3.4.3-2 is required when PCS pressure and temperature conditions are undergoing planned changes. Calculation of the average hourly cooldown rate must consider changes in reactor vessel inlet temperature caused by initiating shutdown cooling, by starting primary coolant pumps with a temperature difference between the steam generator and PCS, or by stopping primary coolant pumps with shutdown cooling in service. The additional restrictions in Figure 3.4.3-2, required for the reactor vessel head nozzle repairs, use the average core exit temperature to provide the best indication available of the temperature of the head inside material temperature. This indication may be either the average of the core exit thermocouples or the vessel outlet temperature.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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

BASES

REFERENCES

1. Safety Evaluation for Palisades Nuclear Plant License Amendment No. 245, dated January 19, 2012

2. 10 CFR 50, Appendix G

3. Deleted

4. ASTM E 185-82, July 1982

5. 10 CFR 50, Appendix H

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

7. Safety Evaluation for Palisades Nuclear Plant License Amendment No. 218, dated November 8, 2004

8. Engineering Analysis EA-EC27959-01, "Palisades Pressure-Temperature Limit Curves and Upper-Shelf Energy Evaluation," February 2012

9. Regulatory Guide 1.99, Revision 2, May 1988

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 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.5.2

This SR requires verification 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES None

INSERT Bases 3.4.6

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.6 PCS Loops - MODE 4

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

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

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

LCO

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

BASES

LCO (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 $\leq 10°F$/hour; or

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

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 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.6.2

This SR requires verification 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.2

This SR requires verification 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

None

INSERT Bases 3.4.9

B 3.4 PRIMARY COOLANT SYSTEMS (PCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

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

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

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

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

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

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

BASES

BACKGROUND (continued)

The requirement to have pressurizer heaters ensures that PCS pressure can be maintained. The pressurizer heaters maintain PCS pressure to keep the primary coolant subcooled. Inability to control PCS pressure during natural circulation flow could result in loss of single phase flow and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensable gases normally present.

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

Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737, "Clarification of TMI Action Plan Requirements," is the reason for their inclusion. The intent is to keep the primary coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power. While a loss of offsite power is a coincident occurrence assumed in the accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated in the accident analyses.

The pressurizer satisfies Criterion 2 (for pressurizer water level) and Criterion 4 (for pressurizer heaters) of 10 CFR 50.36(c)(2).

BASES

LCO

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

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

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

BASES

APPLICABILITY

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

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

ACTIONS

A.1 and A.2

With pressurizer water level not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the plant must be brought to MODE 3, with the reactor tripped, within 6 hours and to MODE 4 within 30 hours. This takes the plant out of the applicable MODES and restores the plant to operation within the bounds of the safety analyses.

Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Further pressure and temperature reduction to MODE 4 brings the plant to a MODE where the LCO is not applicable. The 30 hour time to reach the nonapplicable MODE is reasonable based on operating experience for that evolution.

BASES

ACTIONS (continued)

B.1

If < 375 kW of pressurizer heater capacity is available from either electrical bus 1D or electrical bus 1E, or the pressurizer heaters from electrical bus 1E are not capable of being powered from an emergency power supply, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using the remaining available pressurizer heaters.

C.1

If <375 kW of pressurizer heater capacity is available from both electrical bus 1D and electrical bus 1E, or <375 kW of pressurizer heater capacity is available from electrical bus 1D and the pressurizer heaters from electrical bus 1E are not capable of being powered from an emergency power supply, restoration of either electrical bus pressurizer heaters to an OPERABLE status is required within 24 hours. This Condition is modified by a Note stating it is not applicable if the remaining electrical bus 1D or electrical bus 1E required pressurizer heaters are intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if either electrical bus 1D required pressurizer heaters or electrical bus 1E required pressurizer heaters are discovered to be inoperable, or if both electrical buses' required pressurizer heaters are discovered to be inoperable at the same time. If both electrical buses' required pressurizer heaters are inoperable, pressurizer heater capacity may not be available to maintain subcooling in the PCS loops during natural circulation cooldown following a loss of offsite power. The inoperability of both electrical buses' required pressurizer heaters during the 24 hour Completion Time has been shown to be acceptable based on the infrequent use of the Required Action and the small incremental effect on plant risk (Ref. 3).

BASES

ACTIONS (continued)

D.1 and D.2

If one or more of the electrical buses' required pressurizer heaters cannot be restored to an OPERABLE status within the associated allowed Completion Times, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 30 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 30 hours is reasonable, based on operating experience, to reach MODE 4 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR ensures that during steady state operation, pressurizer water level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. SR 3.4.9.1 is modified by a Note which states that verification of the pressurizer water level is not required to be met until 1 hour after a bubble has been established in the pressurizer and the pressurizer water level has been lowered to its normal operating band. The intent of this Note is to prevent an SR 3.0.4 conflict by delaying the performance of this SR until after the water level in the pressurizer is within its normal operating band following a plant heatup. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the capacity of the associated pressurizer heaters are verified to be ≥ 375 kW. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.9.3

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

This SR confirms that the pressurizer heaters normally fed from electrical bus 1E are capable of being powered from electrical bus 1C by use of a jumper cable. It is not the intent of this SR to physically install the jumper cable, but to verify the necessary components are available for installation and to ensure the procedures and methods used to install the jumper cable are current. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 14

2. FSAR, Section 4.3.7

3. WCAP-16125-NP-A, “Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown,” Revision 2, August 2010.

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

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

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

The lift settings listed in Table 3.4.10-1 correspond to ambient conditions of the valves at nominal operating temperature and pressure. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the PCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit (Ref. 1) could include damage to PCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES

APPLICABLE SAFETY ANALYSES

All accident analyses in the FSAR that require safety valve actuation assume operation of one or more pressurizer safety valves to limit increasing primary coolant pressure. The overpressure protection analysis assumes that the valves open at the high range of the lift setting including the allowable tolerance. The Loss of External Electrical Load incident and Loss of Normal Feedwater Flow incident are the two safety analyses events which rely on the pressurizer safety valves to mitigate an overpressurization of the PCS. The pressurizer safety valves must also accommodate pressurizer in surges that could occur from a Loss of Forced Primary Coolant Flow incident, and a Primary Pump Rotor Seizure incident. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this specification is required to ensure that the accident analysis and design basis calculations remain valid.

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

LCO

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

BASES

APPLICABILITY

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

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

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the PCS overpressure protection system. An inoperable safety valve coincident with an PCS overpressure event could challenge the integrity of the PCPB.

B.1 and B.2

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

INSERT Bases 3.4.10

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

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

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

REFERENCES

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

INSERT Bases 3.4.11

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

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

A motor operated, normally closed, block valve is installed between the pressurizer and each PORV. The function of the block valve is to isolate the PORV. Block valve closure is accomplished manually using controls in the control room and may be used to isolate a leaking PORV to permit continued power operation. Most importantly, the block valve is used to isolate a stuck open PORV to isolate the resulting Loss Of Coolant Accident (LOCA). Closure terminates the PCS depressurization and coolant inventory loss.

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

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

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

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

BASES

BACKGROUND (continued)

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

The PORVs also provide Low Temperature Overpressure Protection (LTOP) during heatup and cooldown. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," addresses this function.

APPLICABLE SAFETY ANALYSES

The PORV small break LOCA break size is bounded by the spectrum of piping breaks analyzed for plant licensing. The possibility of a small break LOCA through the PORV is reduced when the PORV flow path is OPERABLE and the PORV opening setpoint is established to be reasonably remote from expected transient challenges. The possibility is further minimized if the flow path is isolated.

Overpressure protection is provided by safety valves, and analyses do not take credit for the PORV opening for accident mitigation. However, technical findings and regulatory analysis discussed in NUREG-1316, “Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants,” have determined that maintaining the requirements for PORVs and block valves in the technical specifications can increase the reliability of these components and provide assurance they will function as required and that operating experience has shown these components to be important to public health and safety.

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

BASES

LCO

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

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

APPLICABILITY

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

With a PORV in the “AUTO” position in MODES 1 and 2, and MODE 3 with all PCS cold leg temperatures ≥ 430°F, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. A likely cause for PORV small break LOCA is a result of pressure increase transients that cause the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the PCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the steam generators are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODES 1 and 2. Pressure increases are less prominent in MODE 3 with PCS cold leg temperatures < 430°F because the core input energy is reduced, but the PCS pressure is high. Therefore, this LCO is applicable in MODES 1 and 2, and MODE 3 with all PCS cold leg temperatures ≥ 430°F.

The LCO is not applicable in MODE 3 with any PCS cold leg temperatures < 430°F when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODE 3 when any PCS cold leg temperatures are < 430°F, and in MODES 4, 5, and MODE 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

BASES

ACTIONS

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

A.1 and A.2

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

B.1 and B.2

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

BASES

ACTIONS (continued)

B.1 and B.2

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

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

C.1 and C.2

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

D.1 and D.2

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

BASES

ACTIONS (continued)

E.1

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

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

INSERT Bases 3.4.11

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

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

SR 3.4.11.2

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

REFERENCES

None

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

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

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

The potential for vessel overpressurization is most acute when the PCS is water solid, which occurs only while shutdown. Under that condition, a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the PCS P/T limits by a significant amount could cause brittle fracture of the reactor vessel. LCO 3.4.3 requires administrative control of PCS pressure and temperature during heatup and cooldown to prevent exceeding the P/T limits.

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

BASES

BACKGROUND (continued)

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

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

PORV Requirements

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

The LCO presents the PORV setpoints for LTOP by specifying Figure 3.4.12-1, “LTOP Setpoint Limit.” Having the setpoints of both valves within the limits of the LCO ensures the P/T limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases coolant, the system pressure decreases until a reset pressure is reached and the valve closed. The pressure continues to decrease below the reset pressure as the valve closes.

BASES

BACKGROUND (continued)

PCS Vent Requirements

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

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

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

The following vent path examples are acceptable:

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

BASES

BACKGROUND (continued)

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. When originally generated, the validity period for the LTOP Setpoint Limit curve in Figure 3.4.12-1, which is based on the Reference 3 analysis, ended prior to the operating license expiration date. A subsequent analysis was performed (Ref. 9) which demonstrated that the current LTOP Setpoint Limit curve is valid through the operating license expiration date, equivalent to 42.1 effective full power years of operation. Any change to the PCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

Mass Injection Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

Heatup Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of Shutdown Cooling (SDC); or

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

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

Fracture mechanics analyses were used to establish the applicable temperature range for the LTOP LCO as below 430°F. At and above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The pressure-temperature limit curves and LTOP curve are based on reactor vessel material properties which change over time due to radiation embrittlement. These curves are valid for the period of time corresponding to the reactor vessel material condition which was assumed when the curves were generated. At the time the curves were developed, they were based on being valid up to a neutron irradiation accumulation equal to 2.192×10^{19} n (neutrons)/cm^2 (Ref. 3). The vessel materials in the current curve analysis (Ref. 9) were assumed to have a neutron irradiation accumulation equal to 42.1 effective full power years of operation. The current analysis determined an LTOP enable temperature that is bounded by the LTOP LCO.

BASES

APPLICABLE SAFETY ANALYSES (continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the setpoint curve specified in Figure 3.4.12-1 of the accompanying LCO. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient. The valve qualification process considered pressure overshoot and undershoot beyond the PORV opening and closing setpoints, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensure the Reference 1 limits will be met.

The PORV setpoints will be re-evaluated for compliance when the P/T limits are revised. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement caused by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations.

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

PCS Vent Performance

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

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

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

BASES

LCO

This LCO is required to ensure that the LTOP System is OPERABLE. The LTOP System is OPERABLE when both HPSI pumps are incapable of injecting into the PCS and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

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

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

a. Two OPERABLE PORVs; or

b. The PCS depressurized and vented.

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

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

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

BASES

APPLICABILITY

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

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

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

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to inoperable PORVs used for LTOP. There is an increased risk associated with entering MODE 4 from MODE 5 with PORVs used for LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

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

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

BASES

ACTIONS (continued)

B.1

With one required PORV inoperable and pressurizer water level ≤ 57%, the required PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two valves are required to meet the LCO requirement and to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

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

C.1

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

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

BASES

ACTIONS (continued)

D.1

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

The Completion Time of 8 hours to depressurize and vent the PCS is based on the time required to place the plant in this condition and the relatively low probability of an overpressure event during this time period due to operator attention and administrative requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

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

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.2

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

SR 3.4.12.3

The PORV block valve must be verified open to provide the flow path for each required PORV to perform its function when actuated. The valve can be remotely verified open in the main control room.

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.4

A successful CHANNEL FUNCITONAL 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. PORV actuation could depressurize the PCS and is not required. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

A Note has been added indicating this SR is required to be performed 12 hours after decreasing any PCS cold leg temperature to < 430°F. This Note allows a discrete period of time to perform the required test without delaying entry into the MODE of Applicability for LTOP. This option may be exercised in cases where an unplanned shutdown below 430°F is necessary as a result of a Required Action specifying a plant shutdown, or other plant evolutions requiring an expedited cooldown of the plant. The test must be performed within 12 hours after entering the LTOP MODES.

SR 3.4.12.5

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

REFERENCES

1. 10 CFR 50, Appendix G
2. Generic Letter 88-11
3. CPC Engineering Analysis, EA-A-PAL-92-095-01
4. CPC Engineering Analysis, EA-TCD-90-01
5. CPC Engineering Analysis, EA-E-PAL-89-040-1
6. CPC Corrective Action Document, A-PAL-91-011
7. FSAR, Section 7.4
8. Generic Letter 90-06
9. Engineering Analysis EA-EC27959-01, "Palisades Pressure-Temperature Limit Curves and Upper-Shelf Energy Evaluation," February 2012

INSERT Bases 3.4.13

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).

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

INSERT Bases 3.4.13

BASES

APPLICABLE SAFETY ANALYSES (continued)

Atmospheric Dump Valves is the less 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.

The dose consequences resulting from the SGTR, MSLB and CRE accidents are within applicable 10 CFR 50.67 limits and meet 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

LCO (continued)

c. Identified LEAKAGE (continued)

not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

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

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.

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. 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."

SURVEILLANCE REQUIREMENTS (continued)

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.

tA Note under the Frequency column states that this SR is required to be performed during steady state operation. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 7). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

INSERT Bases 3.4.13

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 3.4.14 PCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

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

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

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

BASES

BACKGROUND (continued)

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

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

a. Shutdown Cooling System; and

b. Safety Injection System.

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

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

APPLICABLE SAFETY ANALYSES

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

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

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

INSERT Bases 3.4.14

BASES

LCO

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

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

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

APPLICABILITY

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

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

BASES

ACTIONS

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

A.1 and A.2

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

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

B.1 and B.2

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

BASES

ACTIONS (continued)

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

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

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

Testing is to be performed every 9 months whenever the plant has been in MODE 5 for 7 days or more, but may be extended up to a maximum of 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

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

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

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

SR 3.4.14.2

Verifying that the SDC suction valve interlocks are OPERABLE ensures that PCS pressure will not pressurize the SDC system beyond 125% of its design pressure of 300 psig. The interlock setpoint that prevents the valves from being opened is set so the actual PCS pressure must be < 280 psia to open the valves. This setpoint ensures the SDC design pressure will not be exceeded and the SDC relief valves will not lift. The narrow range pressure transmitters that provide the SDC suction valve interlocks are sensed from the pressurizer. Due to the elevation differences between these narrow range pressure transmitter calibration points and the SDC suction piping, the pressure in the SDC suction piping will be higher than the indicated pressurizer pressure. Due to this pressure difference, the SDC suction valve interlocks are conservatively set at or below 280 psia to ensure that the 300 psig (315 psia) design pressure of the suction piping is not exceeded. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.3

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

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

REFERENCES

1. WASH-1400 (NUREG-75/014), Appendix V, October 1975

2. NUREG-0677, May 1980

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

4. Letter from Consumers Power Company to D.M. Crutchfield (NRC) Requesting a Change to the Palisades Plant Technical Specification, dated July 29, 1982

INSERT Bases 3.4.14

BASES

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

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

INSERT Bases 3.4.15

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.15 PCS Leakage Detection Instrumentation

BASES

BACKGROUND

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

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

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

The primary coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Primary coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. An instrument sensitivity capable of detecting a 100 cm^3/min leak in 45 minutes based on 1% failed fuel is practical for the leakage detection instrument (Ref. 2). Radioactivity detection is included for monitoring gaseous activities because of its sensitivity to PCS LEAKAGE.

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

BASES

BACKGROUND (continued)

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

The containment air cooler design includes a sump with a drain, a liquid level switch, and an overflow path. Normally, very little water will be condensed from the containment atmosphere and the small amount of condensate will easily flow out through the sump drain. If flow to the sump is greater than 20 gpm, the level in the sump will rise to the liquid level switch (approximately 6 inches from the bottom of the sump) and triggers an alarm in the control room. Excessive flow to the sump is indicative of a service water leak, steam leak, or a primary coolant system leak. A steam leak or primary coolant leak would be accompanied by an increase in the containment atmosphere humidity which would be detected by the containment humidity sensors and displayed in the control room. Since excessive containment air cooler drainage may be attributed to causes other than PCS LEAKAGE, an evaluation of PCS LEAKAGE should be confirmed using diverse instrumentation required by this specification.

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

BASES

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system sensitivities are described in the FSAR (Ref. 2). Multiple instrument locations are utilized, if needed, to ensure the transport delay time of the LEAKAGE from its source to an instrument location is acceptable.

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

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

LCO

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

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

APPLICABILITY

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

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

BASES

APPLICABILITY (continued)

Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

If one or two required leak detection instrument channels are inoperable, a periodic surveillance for PCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

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

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

B.1 and B.2

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

BASES

ACTIONS (continued)

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1, SR 3.4.15.2, and SR 3.4.15.3

These SRs require the performance of a CHANNEL CHECK for each required containment sump level indicator, containment atmosphere gaseous activity monitor, and containment atmosphere humidity monitor. The check gives reasonable confidence the channel is operating properly. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.15.4

SR 3.4.15.4 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment air cooler condensate level switch. Since this instrumentation does not include control room indication of flow rate, a CHANNEL CHECK is not possible. The test ensures that the level switch can perform its function in the desired manner. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.15.5, SR 3.4.15.6, and SR 3.4.15.7

These SRs require the performance of a CHANNEL CALIBRATION for each required containment sump level, containment atmosphere gaseous activity, and containment atmosphere humidity channel. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

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

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.16 PCS Specific Activity

BASES

BACKGROUND

10 CFR 100.11 and 10 CFR 50.67 specify the maximum dose an individual at the site boundary can receive for 2 hours during an accident. The limits on specific activity ensure that the doses are held within applicable limits during analyzed transients and accidents.

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

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

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

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the primary coolant ensure that the resulting offsite doses will not exceed applicable limits following a SGTR or other accident. The SGTR safety analysis (Ref. 1) assumes the specific activity of the primary coolant at the LCO limits and an existing primary coolant Steam Generator (SG) tube leakage rate of 0.3 gpm. The analysis also assumes a reactor trip and a turbine trip at the same time as the SGTR event.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

The safety analysis shows the radiological consequences of a SGTR accident are within applicable 10 CFR 50.67 limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limit of 40 μCi/gm for more than 48 hours.

This is acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of a SGTR accident at these permissible levels could increase the site boundary dose levels.

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

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the primary coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy). The limit on DOSE EQUIVALENT I-131 ensures the offsite doses during an accident remains within applicable 10 CFR 50.67 limits.

The SGTR accident analysis (Ref. 1) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in primary coolant radioactivity levels that could, in the event of an SGTR or other accident, lead to site boundary doses that exceed the applicable 10 CFR 50.67 limits.

BASES

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with PCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity is necessary to contain the potential consequences of an SGTR or other accident to within applicable 10 CFR 50.67 limits.

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

ACTIONS

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS.

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate the limit 40 μCi/gm is not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample.

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

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

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

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

INSERT Bases 3.4.16

ACTIONS (continued)

B.1

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

The change within 6 hours to MODE 3 with PCS average temperature < 500°F lowers the saturation pressure of the primary coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is required to reach MODE 3 below 500°F from full power conditions and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

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

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with PCS average temperature at least 500°F. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed to ensure iodine remains within limits during normal operation and following fast power changes when fuel failure is more apt to occur. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency, between 2 hours and 6 hours after any power change of ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results. If any (may be more than one) power change ≥ 15% RTP occurs within a 1 hour period, then more than one sample may be required to ensure that an iodine peak sample is obtained between the 2 and 6 hour Frequency requirement. This SR is modified by a Note which states that the SR is only required to be performed in MODE 1. Entrance into a lower MODE does not preclude completion of this surveillance.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

REFERENCES

1. FSAR, Section 14.15

INSERT Bases 3.4.17

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).

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 that bounds 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 applicable limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3), 10 CFR 50.67 (Ref. 7) 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.

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.

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

ACTIONS (continued)

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.

SURVEILLANCE REQUIREMENTS (continued)

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.

INSERT Bases 3.4.17

BASES

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"
7. 10 CFR 50.67

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Safety Injection Tanks (SITs)

BASES

BACKGROUND

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

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

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

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

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

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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

d. The core is maintained in a coolable geometry.

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

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

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

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

LCO

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

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

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

APPLICABILITY

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

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

BASES

APPLICABILITY (continued)

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

ACTIONS

A.1

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

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

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

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

B.1

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

CE-NPSD-994 (Ref. 3) provides a series of deterministic and probabilistic findings that support the 24 hour Completion Time as having no affect on risk as compared to shorter periods for restoring the SIT to OPERABLE status.

BASES

ACTIONS (continued)

C.1

If the SIT cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power condition in an orderly manner and without challenging plant systems.

D.1

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

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification that each SIT isolation valve is fully open, as indicated in the control room, ensures that SITs are available for injection and ensures timely discovery if a valve should be partially closed. If an isolation valve is not fully open, the rate of injection to the PCS would be reduced. Although a motor operated valve should not change position with power removed, a closed valve could result in not meeting accident analysis assumptions. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.2 and SR 3.5.1.3

SIT borated water volume and nitrogen cover pressure should be verified to be within specified limits in order to ensure adequate injection during a LOCA. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.5

Verification that power is removed from each SIT isolation valve operator ensures that an active failure could not result in the undetected closure of an SIT motor operated isolation valve. If this were to occur, only two SITs would be available for injection, given a single failure coincident with a LOCA. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 14.17

2. FSAR, Chapter 6.1

3. CE-NPSD-994, "CEOG Joint Applications Report for Safety Injection Tank AOT/STI Extension," May 1995

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

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

a. Loss of Coolant Accident (LOCA);

b. Control Rod Ejection accident;

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

d. Steam Generator Tube Rupture (SGTR).

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

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

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

BACKGROUND (continued)

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

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

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

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

BACKGROUND (continued)

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

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

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

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

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

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

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

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

APPLICABLE SAFETY ANALYSES (continued)

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

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

LCO

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

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

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

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

APPLICABILITY

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

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

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

ACTIONS

A.1

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

ACTIONS
(continued)

B.1

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

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

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

C.1 and C.2

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

If the inoperable ECCS trains cannot be restored to OPERABLE status within the required Completion Times of Condition A and B, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and PCS temperature reduce to < 325°F within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power in an orderly manner and without challenging plant systems.

ACTIONS (continued)

D.1

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

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

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

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

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

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

Placing HS-3027A and HS-3027B for CV-3027, and HS-3056A and HS-3056B for CV-3056, in the open position ensures that the valves cannot be inadvertently misaligned or change position as the result of an active failure. These valves are of the type described in Reference 4, which can disable the function of both ECCS trains and invalidate the accident analysis. CV-3027 and CV-3056 are capable of being closed from the control room since the SIRWT must be isolated from the containment during the recirculation phase of a LOCA. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.2

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.5.2.3

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.4

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

SR 3.5.2.5, SR 3.5.2.6, and SR 3.5.2.7

These SRs demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated actuation signal, i.e., on an SIS or RAS, that each ECCS pump starts on receipt of an actual or simulated actuation signal, i.e., on an SIS, and that the LPSI pumps stop on receipt of an actual or simulated actuation signal, i.e., on an RAS. RAS opens the HPSI subcooling valve CV-3071, if the associated HPSI pump is operating. After the containment sump valve CV-3030 opens from RAS, HPSI subcooling valve CV-3070 will open, if the associated HPSI pump is operating. RAS will re-position CV-3001 and CV-3002 to a predetermined throttled position. RAS will close

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5, SR 3.5.2.6, and SR 3.5.2.7

containment spray valve CV-3001, if containment sump valve CV-3030 does not open. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The actuation logic is tested as part of the Engineered Safety Feature (ESF) testing, and equipment performance is monitored as part of the INSERVICE TESTING PROGRAM.

SR 3.5.2.8

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

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

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.9

Periodic inspection of the ECCS containment sump passive strainer assemblies ensures that the post-LOCA recirculation flowpath to the ECCS train containment sump suction inlets is unrestricted. Periodic inspection of the containment sump entrance pathways, which include containment sump passive strainer assemblies, containment sump downcomer debris screens, containment floor drain debris screens, containment sump vent debris screens, and reactor cavity corium plug bottom cup support assemblies, ensures that the containment sump stays in proper operating condition. The migration of LOCA-generated debris larger than the strainer perforation diameter through the two one-inch reactor cavity drain line corium plugs is not considered to be credible. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 5.1

2. FSAR, Section 14.17

3. NRC Memorandum to V. Stello, Jr., from R. L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975

4. IE Information Notice No. 87-01, January 6, 1987

5. CE-NPSD-994, "CEOG Joint Applications Report for Safety Injection Tank AOT/STI Extension," May 1995

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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

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

BASES

LCO

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

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

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

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

APPLICABILITY

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

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

BASES

APPLICABILITY (continued)	In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "PCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "PCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."
ACTIONS	A.1 With no LPSI train OPERABLE, the plant is not prepared to respond to a loss of coolant accident. Action must be initiated Immediately to restore at least one LPSI train to OPERABLE status. The Immediate Completion Time reflects the importance of maintaining an OPERABLE LPSI train and ensures that prompt action is taken to restore the required cooling capacity.
SURVEILLANCE REQUIREMENTS	SR 3.5.3.1 The applicable Surveillance descriptions from Bases 3.5.2 apply.
REFERENCES	The applicable references from Bases 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Safety Injection Refueling Water Tank (SIRWT)

BASES

BACKGROUND

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

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

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

BASES

BACKGROUND (continued)

This LCO ensures that:

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

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

c. The reactor remains subcritical following a LOCA.

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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

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

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

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

BASES

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

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

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

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

BASES

LCO

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

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

APPLICABILITY

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

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

ACTIONS

A.1

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

BASES

ACTIONS (continued)

B.1

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

C.1 and C.2

If the SIRWT cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.4.2 and SR 3.5.4.3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.4.2 and SR 3.5.4.3 (continued)

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

SR 3.5.4.4

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Containment Sump Buffering Agent and Weight Requirements

BASES

BACKGROUND

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

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

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

BASES

BACKGROUND (continued)

Adjusting the pH of the recirculation solution to levels above 7.0 prevents a significant fraction of the dissolved iodine from converting to a volatile form. The higher pH thus decreases the level of airborne iodine in containment and reduces the radiological consequences from containment atmosphere leakage following a LOCA. Maintaining the solution pH above 7.0 also reduces the occurrence of SCC of austenitic stainless steel components in containment. Reducing SCC reduces the probability of failure of components.

The highest hydrated form of STB (decahydrate sodium tetraborate) is used to inhibit the absorption of large amounts of water from the humid atmosphere. Thus, it will undergo less physical and chemical change than the anhydrous form of STB.

APPLICABLE SAFETY ANALYSES

The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being ≥ 7.0. The radionuclide releases from the containment atmosphere and the consequences of a LOCA would be increased if the pH of the recirculation water were not adjusted to 7.0 or above.

The containment hydrogen concentration analysis used in the evaluation of the Maximum Hypothetical Accident (MHA) assumes the pH of the containment sump water is between 7.0 and 8.0. The acceptance criteria of the MHA includes a containment lower flammability limit of 4 volume percent for hydrogen. Containment sump water with a pH greater than 8.0 could result in excess hydrogen generation in containment and invalidate the conclusions of the MHA.

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

LCO

The quantity of STB placed in containment is designed to adjust the pH of the sump water to be between 7.0 and 8.0 after a LOCA. A pH > 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH > 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

BASES

LCO (continued)

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

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

The STB is stored in wire mesh baskets placed inside the containment at the 590 ft elevation. Any quantity of STB between 8,186 and 10,553 lb. will result in a pH in the desired range.

APPLICABILITY

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

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

ACTIONS

A.1

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

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

B.1 and B.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Periodic determination of the mass of STB in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the STB during normal operation. This is required to determine that ≥ 8,186 lbs and ≤ 10,553 lbs of equivalent weight of decahydrate STB are contained in the STB baskets. In the event that the total STB weight is less than the minimum weight, a chemical test is performed to confirm that the weight change is due to the dehydration of the decahydrate form of the STB. It is not necessary to replenish STB if the minimum weight is not met solely due to dehydration of the material. This requirement ensures that there is an adequate mass of STB to adjust the pH of the post LOCA sump solution to a value ≥ 7.0 and ≤ 8.0.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.5.2

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

Adequate buffering capability is verified by a measured pH of the sample STB in boric acid solution. The quantity of the STB sample and quantity and boron concentration of the water are chosen to be representative of post-LOCA conditions. The pH is measured at 25°C and is verified to be between 7.0 and 8.0.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 6.4

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

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

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

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

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

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

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

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

BACKGROUND (continued)

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";

c. The equipment hatch is properly closed and sealed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

A Loss of Coolant Accident (LOCA) and a control rod ejection accident are the two DBAs that are analyzed for release of fission products within containment (Ref. 1). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.10% of containment air weight per day at a design pressure of 55 psig and a design temperature of 283°F (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

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

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Containment Leak Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

Technical Specification ADMIN 5.5.14 defines L_a as the maximum allowable leakage rate at pressure P_a. The P_a value of 54.2 psig represents the analytical value for a large break LOCA found in Reference 1.

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

LCO (continued)

Individual leakage rates that may be specified for the containment air lock (LCO 3.6.2) and purge valves which have resilient seals (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

APPLICABILITY

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

ACTIONS

A.1

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

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

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

SR 3.6.1.2

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

REFERENCES

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

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

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

Air lock integrity and leak tightness are essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the plant safety analysis.

APPLICABLE SAFETY ANALYSES

A Loss of Coolant Accident (LOCA) and a control rod ejection accident are the two DBAs that are analyzed for release of fission products within containment (Ref. 1). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.10% of containment air weight per day at a design pressure of 55 psig and a design temperature of 283°F (Ref. 2). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

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

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to limiting the containment leakage rate to $\leq 1.0\ L_a$. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Technical Specification ADMIN 5.5.14 defines L_a as the maximum allowable leakage rate at pressure P_a. The P_a value of 54.2 psig represents the analytical value for a large break LOCA found in Reference 1.

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

APPLICABILITY

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

ACTIONS

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

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

ACTIONS (continued)

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

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

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

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

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

ACTIONS

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

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

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

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

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

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

ACTIONS

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

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

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

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

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

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

ACTIONS (continued)

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

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

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit into and out of containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the airlock is used for entry and exit (procedures require strict adherence to single door opening).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 14

2. FSAR, Section 5.8

3. 10 CFR 50, Appendix J, Option B

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

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

The Containment Isolation System is designed to provide isolation capability following a Design Basis Accident (DBA) for fluid lines that penetrate containment. Major nonessential lines (i.e., fluid systems that do not perform an immediate accident mitigation function) that penetrate containment, except for the main steam lines and instrument air line, are either automatically isolated following an accident or are normally maintained closed in MODES 1, 2, 3, and 4. Containment isolation occurs upon receipt of a Containment High Pressure (CHP) signal or a Containment High Radiation (CHR) signal. However, not all containment isolation valves are actuated by both signals. The signals close automatic containment isolation valves in fluid penetrations that are required to be isolated during accident conditions in order to minimize release of fission products from the Primary Coolant System (PCS) to the environment. Other penetrations that are required to be isolated during accident conditions are isolated by the use of valves or check valves in the closed position, or blind flanges. As a result, the containment isolation devices help ensure that the containment atmosphere will be isolated in the event of a release of fission products to the containment atmosphere from the PCS following a DBA.

BASES

BACKGROUND

The plant safety analyses (Reference 5) assume containment isolation for the mitigation of a Loss of Coolant Accident (LOCA) and a control rod ejection accident. The Main Steam Line Break, Steam Generator Tube Rupture, and Control Rod Ejection accident analyses include scenarios in which the mass of steam from the Steam Generator is assumed to be released directly to the environment, and no credit is taken for containment isolation to mitigate the radiological consequences of those accidents. For other analyzed accidents, a release path via fluid lines connected directly to the secondary side of the steam generators would require a passive failure, and Palisades is not required to postulate passive failures of equipment performing safety functions in accident scenarios (Reference 6). Therefore, valves in fluid lines connected directly to the secondary side of the steam generators are not included in this Technical Specification.

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

The 8 inch purge exhaust valves are designed for purging the containment atmosphere to the stack while introducing filtered makeup, through the 12 inch air room supply valves from the outside, when the plant is shut down during refueling operations and maintenance. The purge exhaust valves and air room supply valves are air operated isolation valves located outside the containment. These valves are operated manually from the control room. These valves will close automatically upon receipt of a CHP or CHR signal. The air operated valves fail closed upon a loss of air. These valves are not qualified for automatic closure from their open position under DBA conditions. Therefore, these valves are locked closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Open purge exhaust or air room supply valves, following an accident that releases contamination to the containment atmosphere, would cause a significant increase in the containment leakage rate.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the release of fission products from the primary coolant system to the environment, and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve (device) OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

BASES

APPLICABLE SAFETY ANALYSES (continued)

A Loss of Coolant Accident (LOCA) and a control rod ejection accident are the two DBAs that require isolation of containment to minimize release of fission products to the environment (Ref. 5). In the analysis for each of these accidents, it is assumed that containment isolation devices that are required to be closed during accident conditions are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation devices (including containment purge valves) are minimized. The safety analysis assumes that the purge exhaust and air room supply valves are closed at event initiation.

The DBA analysis assumes that, within 25 seconds after receiving a CHP or CHR signal each automatic power operated valve is closed and containment leakage terminated except for the design leakage rate.

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

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

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

LCO

Containment isolation valves form a part of the containment boundary. Compliance with this LCO will ensure a containment configuration that will limit leakage to those leakage rates assumed in the safety analyses. Containment penetrations for fluid systems that perform an accident mitigation function are not required to be isolated.

LCO (continued)

Containment isolation valves (devices) consist of isolation valves (manual valves, check valves, air operated valves, and motor operated valves), and blind flanges. There are two major categories of containment isolation devices that are used depending on the type of penetration and the function of the associated piping system:

a. Active - automatic containment isolation devices that, following an accident, either receive a containment isolation signal to close, or close as a result of differential pressure;

b. Passive - normally closed containment isolation devices that are maintained closed in MODES 1, 2, 3, and 4 since they do not receive a containment isolation signal to close and the penetration is not used for normal power operation.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate upon receipt of a CHP or CHR signal as appropriate. Check valves are verified to be OPERABLE through the valve INSERVICE TESTING PROGRAM. The purge exhaust and air room supply valves must be locked closed.

The normally closed isolation devices are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, check valves are closed with flow secured through the pipe, or blind flanges are in place.

The devices covered by this LCO are listed in the FSAR (Ref. 2).

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

This LCO provides assurance that the containment isolation devices will perform their designed safety functions to minimize the release of fission products from the primary coolant system to the environment and establish the containment boundary during accidents.

BASES

APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of fission products to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by four notes. Note 1 allows isolated penetration flow paths, except for 8 inch exhaust and 12 inch air room supply purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the device controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the fact that the 8 inch purge exhaust valves and the 12 inch air room supply valves may be unable to close in the environment following a LOCA and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, these valves may not be opened under administrative controls.

Note 2 provides clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation device. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation devices are governed by subsequent Condition entry and application of associated Required Actions.

Note 3 ensures that appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation device.

Note 4 requires entry into the applicable Conditions and Required Actions of LCO 3.6.1 when leakage results in exceeding the overall containment leakage limit.

A.1 and A.2

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

ACTIONS

A.1 and A.2 (continued)

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

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

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

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

ACTIONS

A.1 and A.2 (continued)

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

B.1

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

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

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

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

ACTIONS
(continued)

C.1 and C.2

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

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

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

BASES

ACTIONS

C.1 and C.2 (continued)

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

D.1

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

E.1 and E.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2

This SR requires verification that each manual containment isolation valve and blind flange located outside containment, and not locked, sealed, or otherwise secured in position, and required to be closed during accident conditions, is closed. The SR helps to ensure that post accident leakage of fission products outside the containment boundary is within design limits. This SR does not require any testing or device manipulation. Rather, it involves verification that those containment isolation devices outside containment and capable of being mispositioned are in the correct position. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to devices that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

SR 3.6.3.3

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3 (continued)

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

SR 3.6.3.4

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

SR 3.6.3.5

For containment 8 inch purge exhaust and 12 inch air room supply valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B (Ref. 3), is required to ensure the valves are physically closed (SR 3.6.3.1 verifies the valves are locked closed). Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to minimize leakage of fission products from containment following a DBA. This SR ensures each automatic containment isolation valve will actuate to its isolation position on an actual or simulated actuation signal, i.e., CHP or CHR. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 5.8
2. FSAR, Section 6.7.2 and Table 6-14
3. 10 CFR 50, Appendix J, Option B
4. Deleted
5. FSAR, Chapter 14
6. FSAR, Section 1.4.16

INSERT Bases 3.6.4

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered for determining the maximum containment internal pressure are the LOCA and MSLB. A large break LOCA results in the highest calculated internal containment pressure of 54.2 psig, which is below the internal design pressure of 55 psig. The postulated DBAs are analyzed assuming degraded containment Engineered Safety Feature (ESF) systems (i.e., assuming the limiting single active failure). See the Bases for 3.6.1, “Containment,” for a discussion on containment pressures resulting from a LOCA.

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

LCO

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

APPLICABILITY

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

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

ACTIONS

A.1

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

BASES

ACTIONS (continued)

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that operation remains within the limits assumed in the accident analyses. The limit of 1.0 psig for MODES 1 and 2, 1.5 psig for MODES 3 and 4 are the actual limits used in the accident analysis and do not account for instrument inaccuracies. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 14.18

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analysis for containment. The accident analyses and evaluations considered both LOCAs and MSLBs for determining the maximum peak containment pressures and temperatures. The LOCA event is bounding with respect to peak containment pressure, and the MSLB event is bounding with respect to peak containment temperature. This is due to the differences in the magnitude and timing of the mass and energy release rates between the two events. The LOCA peak pressure occurs prior to any containment heat removal components being placed in service. The MSLB peak temperature occurs after heat removal equipment has been in operation.

The initial pre-accident temperature inside containment was assumed to be 145°F to provide analysis margin from the Technical Specification limit of 140°F (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

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

APPLICABILITY

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

ACTIONS

A.1

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

BASES

ACTIONS (continued)

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. The 145°F limit is the actual limit assumed for the accident analyses and does not account for instrument inaccuracies. Instrument uncertainties are accounted for in the surveillance procedure. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 5.8

2. FSAR, Section 14.18

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Cooling Systems

BASES

BACKGROUND

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

The Containment Air Cooler System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The systems are arranged with two spray pumps powered from one diesel generator, and with one spray pump and three air cooler fans powered from the other diesel generator. The Containment Spray System was originally designed to be redundant to the Containment Air Coolers (CACs) and fans. These systems were originally designed such that either two containment spray pumps or three CACs could limit containment pressure to less than design. However, the current safety analyses take credit for one containment spray pump when evaluating cases with three CACs, and no air cooler fans in cases with two spray pumps and both Main Steam Isolation Valve (MSIV) bypass valves closed. If an MSIV bypass valve is open, 2 service water pumps and 2 CACs are also required to be OPERABLE in addition to the 2 spray pumps for containment heat removal.

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

BASES

BACKGROUND (continued)

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

Containment Spray System

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

Normally, both Shutdown Cooling Heat Exchangers must be available to provide cooling of the containment spray flow in the event of a Loss of Coolant Accident. If the Containment Spray side (tube side) of one SDC Heat Exchanger is out of service, 100% of the required post accident cooling capability can be provided, if other equipment outages are limited (refer to Bases for Required Action C.1).

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

BASES

BACKGROUND

Containment Spray System (continued)

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

The Containment Spray System is actuated either automatically by a Containment High Pressure (CHP) signal or manually. An automatic actuation opens the containment spray header isolation valves, starts the three containment spray pumps, and begins the injection phase. Individual component controls may be used to manually initiate Containment Spray. The injection phase continues until an SIRWT Level Low signal is received. The Low Level signal for the SIRWT generates a Recirculation Actuation Signal (RAS) that aligns valves from the containment spray pump suction to the containment sump. RAS re-positions CV-3001 and CV-3002 to a predetermined throttled position to ensure adequate containment spray pump NPSH. RAS opens the HPSI subcooling valve CV-3071, if the associated HPSI pump is operating. After the containment sump valve CV-3030 opens from RAS, HPSI subcooling valve CV-3070 will open, if the associated HPSI pump is operating. RAS will close containment spray valve CV-3001, if containment sump valve CV-3030 does not open. The Containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

The containment spray pumps also provide a required support function for the High Pressure Safety Injection pumps as described in the Bases for specification 3.5.2. The High Pressure Safety Injection pumps alone may not have adequate NPSH after a postulated accident and the realignment of their suctions from the SIRWT to the containment sump. Flow is automatically provided from the discharge of the containment spray pumps to the suction of the High Pressure Safety Injection (HPSI) pumps after the change to recirculation mode has occurred, if the HPSI pump is operating. The additional suction pressure ensures that adequate NPSH is available for the High Pressure Safety Injection pumps.

BASES

BACKGROUND (continued)

Containment Air Cooler System

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

The DG which powers the fans associated with VHX-1, VHX-2, and VHX-3 (V-1A, V-2A and V-3A) also powers two service water pumps. This is necessary because if reliance is placed solely on the train with one spray pump and three CACs, at least two service water pumps must be OPERABLE to provide the necessary service water flow to assure OPERABILITY of the CACs.

Each CAC has two vaneaxial fans with direct connected motors which draw air through the cooling coils. Both of these fans are normally in operation, but only one fan and motor for each CAC is rated for post accident conditions. The post accident rated “safety related” fan units, V-1A, V-2A, and V-3A, serve to provide forced flow for the associated cooler. A single operating safety related spray header will provide enough air flow to assure that there is adequate mixing of unsprayed containment areas to assure the assumed iodine removal by the containment spray. In post accident operation following a SIS, all four Containment air coolers are designed to change automatically to the emergency mode.

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

BASES

APPLICABLE SAFETY ANALYSES

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

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

1. A Large Break LOCA concurrent with a loss of offsite power,

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

3. An MSLB occurring at 0% RTP with both MSIV bypass valves open, both with and without offsite power available.

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

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

The MSLB event is analyzed at various power levels with both MSIV bypass valves closed, and at 0% RTP (MODE 2) with both MSIV bypass valves open. Having any MSIV bypass valve open allows additional blowdown from the intact steam generator. These cases consider single active failure scenarios both with and without offsite power available. With offsite power available, the analysis evaluates failure of various relays responsible for starting containment heat removal components on receipt of SIS or CHP signals. On loss of offsite power, the analysis evaluates failure of an emergency diesel generator resulting in one train of containment cooling being rendered inoperable. Generally, cases with offsite power available are bounding as the primary coolant pumps remain in service resulting in forced convection through the steam generators increasing the blowdown energy.

BASES

APPLICABLE ANALYSES (continued)

The analysis and evaluation show that under the worst-case scenario, the highest peak containment pressure and the peak containment vapor temperature are within the design basis. (See the Bases for Specifications 3.6.4, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations considered a range of power levels and equipment configurations as described in Reference 2. The peak containment pressure case is the large break LOCA with initial (pre-accident) conditions of 145°F and 15.7 psia. The peak temperature case is the 0% power MSLB with initial (pre-accident) conditions of 145°F and 16.2 psia. The analyses also assume a response time delayed initiation in order to provide conservative peak calculated containment pressure and temperature responses.

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

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

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

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

LCO

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

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

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

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

APPLICABILITY

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

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

BASES

ACTIONS

A.1

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

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

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

B.1 and B.2

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

If the inoperable containment cooling trains cannot be restored to OPERABLE status within the required Completion Time of Condition A, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

C.1

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

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

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

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

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

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

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

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

The components described in items 1 and 2 directly above, are necessary to mitigate a MSLB where offsite power is available and primary coolant pumps continue to operate. Therefore, components from both trains of containment heat removal are required.

BASES

ACTIONS

C.1 (continued)

If the Containment Spray side (tube side) of SDC Heat Exchanger E-60B is out of service, 100% of the required post accident cooling capability can be provided, if other equipment outages are limited. One hundred percent of the post accident cooling can be provided with the Containment Spray side of SDC Heat Exchanger E-60B out of service if the following equipment is OPERABLE: three safety related Containment Air Coolers, two Containment Spray Pumps, two spray headers, CCW pumps P-52A and P-52B, two SWS pumps, and both CCW Heat Exchangers, and if

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

2. Two CCW isolation valves for the non-safety related loads outside the containment, CV-0944A and CV-0944 (or CV-0977B), are OPERABLE.

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

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

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

SR 3.6.6.2

Operating each safety related Containment Air Cooler fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and are functioning properly. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.3

Verifying the containment spray header is full of water to the 735 ft elevation minimizes the time required to fill the header. This ensures that spray flow will be admitted to the containment atmosphere within the time frame assumed in the containment analysis. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.4

Verifying a total service water flow rate of ≥ 4800 gpm to CACs VHX-1, VHX-2, and VHX-3, when aligned for accident conditions, provides assurance the design flow rate assumed in the safety analyses will be achieved (Ref. 8). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.5

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.6 and SR 3.6.6.7

SR 3.6.6.6 verifies each automatic containment spray valve actuates to its correct position upon receipt of an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. SR 3.6.6.7 verifies each containment spray pump starts automatically on an actual or simulated actuation signal. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

SR 3.6.6.8

This SR verifies each safety related containment cooling fan actuates upon receipt of an actual or simulated actuation signal. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.9

With the containment spray inlet valves closed and the spray header drained of any solution, an inspection of spray nozzles, or a test that blows low-pressure air or smoke through test connections can be completed. Performance of this SR demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Verification following maintenance which could result in nozzle blockage is appropriate because this is the only activity that could lead to nozzle blockage.

BASES

REFERENCES

1. FSAR, Section 5.1

2. FSAR, Section 14.18

3. FSAR, Sections 6.2

4. FSAR, Section 6.3

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

6. FSAR, Table 14.18.1-3

7. FSAR, Table 14.18.2-1

8. FSAR, Table 9-1

9. EA-GOTHIC-04-09 Rev. 3, Containment Response to a MSLB Using GOTHIC 7.2a, October 2010.

10. EA-GOTHIC-04-08, Rev. 3, Containment Response to a LOCA Using GOTHIC 7.2a, October 2010.

LCO 3.6.7 AND BASES DELETED: REFER TO AMENDMENT 221 DATED 1/11/05

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND	The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the Primary Coolant Pressure Boundary (PCPB) by providing a heat sink for the removal of energy from the Primary Coolant System (PCS) if the preferred heat sink, provided by the condenser and Circulating Water System, is not available. Twelve MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 4.3.4 (Ref. 1). The MSSV rated capacity passes the full steam flow at RTP plus instrument error with twenty-three valves full open. This meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III (Ref. 2). The MSSV design includes staggered lift settings, according to Table 3.7.1-1, in the accompanying LCO, so that only the number of valves needed will actuate. Staggered lift settings reduce the potential for valve chattering because of insufficient steam pressure to fully open all valves following a turbine reactor trip.
APPLICABLE SAFETY ANALYSES	The design basis for the MSSVs comes from Reference 1; the purpose is to limit secondary system pressure to $\leq$ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any Anticipated Operational Occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis. (As defined in 10 CFR 50, Appendix A, "Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.")

BASES

APPLICABLE SAFETY ANALYSES (continued)

The events that challenge the MSSV relieving capacity, and thus PCS pressure, are those characterized as decreased heat removal events, and are presented in the FSAR, Sections 14.12 and 14.13 (Refs. 3 and 4) respectfully. Of these, the full power loss of external load event is the most limiting. The event is initiated by either a loss of external electrical load or a turbine trip. No credit is taken for direct reactor trip on turbine trip, the turbine bypass valve, atmospheric dump valves, or pressurizer PORVs. The reduced heat transfer causes an increase in PCS temperature, and the resulting PCS fluid expansion causes an increase in pressure. The PCS pressure increases to ≤ 2614.9 psia, this peak pressure is < 110% of the design pressure, or 2750 psia for the primary system, with the pressurizer safety valves providing relief capacity. The secondary system pressure increases to 1040.8 psia, this pressure is < 110% of the design pressure, or 1100 psia for the secondary system, with the MSSVs providing relief capability.

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

LCO

This LCO requires twenty-three MSSVs to be OPERABLE in compliance with Reference 2. The OPERABILITY of the MSSVs is defined as the ability to open within the lift setting tolerances and to relieve steam generator overpressure. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the INSERVICE TESTING PROGRAM.

The lift settings, according to Table 3.7.1-1 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the PCPB.

APPLICABILITY

In MODES 1, 2, and 3 a minimum of twenty-three MSSVs are required to be OPERABLE, to provide overpressure protection required by both ASME Code and the accident analysis.

In MODES 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

BASES

ACTIONS

A.1

With one or more required MSSVs inoperable, the ability to limit system pressure during accident conditions will be degraded. The four hour Completion Time allows the operator a reasonable amount of time to make minor repairs or adjustments to restore the required number of inoperable MSSVs to OPERABLE status.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the INSERVICE TESTING PROGRAM. The safety and relief valve tests are performed in accordance with ASME Code (Ref. 5) and include the following for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements.

Table 3.7.1-1 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

The ambient temperature of the operating environment shall be simulated during the set-pressure test in accordance with Reference 5.

REFERENCES

1. FSAR, Section 4.3.4
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components
3. FSAR, Section 14.12
4. FSAR, Section 14.13
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

INSERT Bases 3.7.2

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a High Energy Line Break (HELB) downstream of the MSIV. MSIV closure terminates flow from the unaffected (intact) steam generator for breaks upstream of the other MSIV.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the Main Steam Safety Valves (MSSVs), atmospheric dump valves, and auxiliary feedwater pump turbine steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, turbine bypass valve, and other auxiliary steam supplies from the steam generators, assuming the normally closed MSIV bypass valves are closed. The MSIV bypass valves do not receive an isolation signal and might be open during zero power conditions.

The MSIVs close on isolation signals generated by either Steam Generator Low Pressure or Containment High Pressure. The MSIVs fail closed on loss of air. The isolation signal also actuates the Main Feedwater Regulating Valves (MFRVs) and MFRV bypass valves to close. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the Main Steam Line Break (MSLB) inside containment, as discussed in the FSAR, Section 14.18 (Ref. 2). It is also influenced by the accident analysis of the MSLB events presented in the FSAR, Section 14.14 (Ref. 3). The MSIVs are swing disc check valves. The inherent characteristic of this type of valve allows for reverse flow through the valve on a differential pressure even if the valve is closed. In the event of an MSLB, if the MSIV associated with the unaffected steam generator fails to close, both steam generators may blowdown. This failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a double steam generator blowdown event, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

There are three different limiting MSLB cases that have been evaluated, one for fuel integrity and two for containment analysis (one for containment temperature and one for containment pressure). The limiting case for containment temperature is the hot full power MSLB inside containment following a turbine trip. At hot full power, the stored energy in the primary coolant is maximized.

The limiting case for the containment analysis for containment pressure and fuel integrity is the hot zero power MSLB inside containment. At zero power, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Reverse flow due to the open MSIV bypass valves, contributes to the total release of the additional mass and energy. With the most reactive control rod assumed stuck in the fully withdrawn position, there is an increased possibility that the core will return to power. The core is ultimately shut down by a combination of doppler feedback, steam generator dryout, and borated water injection delivered by the Emergency Core Cooling System.

The accident analysis compares several different MSLB events against different acceptance criteria. The MSLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The MSLB inside containment at hot full power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following a turbine trip.

With offsite power available, the primary coolant pumps continue to circulate coolant through the steam generators, maximizing the Primary Coolant System (PCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Safety Injection (HPSI) pumps, is delayed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An MSLB inside containment. For this accident scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator.

b. A break outside of containment and upstream from the MSIVs. This scenario is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled PCS cooldown and positive reactivity addition. Closure of the MSIVs limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs. This type of break will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the turbine bypass valve will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIVs isolates the affected steam generator from the intact steam generator and minimizes radiological releases.

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

LCO

This LCO requires that the MSIV in each of the two steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures in excess of 10 CFR 50.67 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when both MSIVs are closed and deactivated when there is significant mass and energy in the PCS and steam generators. When the MSIVs are closed, they are already performing their safety function. Deactivation can be accomplished by the removal of the motive force (e.g., air) to the valve to prevent valve opening.

INSERT Bases 3.7.2

BASES

APPLICABILITY (continued)

In MODE 4, the steam generator energy is low; therefore, the MSIVs are not required to be OPERABLE.

In MODES 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the plant hot. The 8 hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment.

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, 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 MODE 2 within 6 hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2 in an orderly manner and without challenging plant systems.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is consistent with that allowed in Condition A.

BASES

ACTIONS (continued)

C.1 and C.2 (continued)

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid.

The once per 7 days Completion Time is reasonable, based on engineering judgment, MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position. 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 3.7.2 C.2 must be initially performed within 7 days without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or closed, 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 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from MODE 2 in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is ≤ 5.0 seconds on an actual or simulated actuation signal from each train under no flow conditions. Specific signals (e.g., Containment High Pressure, Steam Generator Low Pressure, handswitch) are tested under Section 3.3, “Instrumentation.” The MSIV closure time is assumed in the MSLB and containment analyses. This SR is normally performed during a refueling outage. The MSIVs are not tested at power since even a part stroke exercise increases the risk of a valve closure with the plant generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5) requirements during operation in MODES 1 and 2.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 10.2
2. FSAR, Section 14.18
3. FSAR, Section 14.14
4. 10 CFR 50.67
5. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

BASES

BACKGROUND

The MFRVs and MFRV bypass valves in conjunction with feed pump speed, control Main Feedwater (MFW) flow to the steam generators for level control during normal plant operation. The valves also isolate MFW flow to the secondary side of the steam generators following a High Energy Line Break (HELB). Closure of the MFRVs and MFRV bypass valves terminates flow to both steam generators. Closure of the MFRV and MFRV bypass valve effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for Main Steam Line Breaks (MSLBs) inside containment, and reducing the cooldown effects.

The MFRVs and MFRV bypass valves isolate MFW in the event of a secondary side pipe rupture inside containment to limit the quantity of high energy fluid that enters containment through the break. Controlled addition of Auxiliary Feedwater (AFW) is provided by a separate piping system.

One MFRV and one MFRV bypass valve are located on each MFW line outside containment. The piping volume from the valves to the steam generator must be accounted for in calculating mass and energy releases following an MSLB.

The MFRVs and MFRV bypass valves close on receipt of an isolation signal generated by either; steam generator low pressure from its respective steam generator, or containment high pressure. These isolation signals also actuate the Main Steam Isolation Valves (MSIVs) to close. The MFRVs and MFRV bypass valves may also be actuated manually. The MFRVs and MFRV Bypass valves are non-safety grade valves located on non-safety grade piping that fail “as-is” on a loss of air. If required, MFW isolation can be accomplished using manually operated valves upstream or downstream of the MFRVs and MFRV Bypass valves. In addition, each MRFV is equipped with a handwheel that can be used to isolate this MFW flowpath.

A description of the MFRVs and MFRV bypass valves is found in the FSAR, Section 10.2.3 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

Closure of the MFRVs is an assumption in the MSLB containment response analysis. Closure of the MFRVs and MFRV bypass valves is also assumed in the MSLB core response (DNB) analysis.

Failure of an MFRV or MFRV bypass valve to close following an MSLB can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an MSLB event. However, this failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a single steam generator blowdown event with continued feedwater, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

The MFRVs and MFRV bypass valves satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO ensures that the MFRVs and MFRV bypass valves will isolate MFW flow to the steam generators following an MSLB. This LCO requires that both MFRVs and both MFRV bypass valves be OPERABLE. The MFRVs and MFRV bypass valves are considered OPERABLE when the isolation times are within limits, and are closed on an isolation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an MSLB inside containment.

BASES

APPLICABILITY

All MFRVs and MFRV bypass valves must be OPERABLE, or either closed and deactivated, or isolated by closed manually actuated valves, whenever there is significant mass and energy in the Primary Coolant System and steam generators.

In MODES 1, 2, and 3, the MFRVs or MFRV bypass valves are required to be OPERABLE, except when both MFRVs and both MFRV bypass valves are either closed and deactivated, or isolated by closed manually actuated valves, in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are either closed and deactivated, or isolated by closed manually actuated valves, they are already performing their safety function.

Once the valves are closed, deactivation can be accomplished by the removal of the motive force (e.g., electrical power, air) to the valve to prevent valve opening. Closing another manual valve in the flow path either remotely (i.e., control room hand switch) or locally by manual operation satisfies isolation requirements.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFRVs and MFRV bypass valves are not required to be OPERABLE.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one or more MFRV or MFRV bypass valve inoperable, action must be taken to close or isolate the inoperable valve(s) within 8 hours. When these valve(s) are closed or isolated, they are performing their required safety function (e.g., to isolate the line).

The 8 hour Completion Time is reasonable to close the MFRV or MFRV bypass valve, which includes performing a controlled plant shutdown to a condition that supports isolation of the affected valve(s). 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.

BASES

ACTIONS (continued)

A.1 and A.2 (continued)

Therefore, while Required Action 3.7.3 A.2 must be initially performed within 7 days without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

B.1 and B.2

If the MFRVs or MFRV bypass valves cannot be restored to OPERABLE status, closed, or isolated in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies the closure time for each MFRV and MFRV bypass valve is ≤ 22.0 seconds on an actual or simulated actuation signal. Specific signals (e.g., steam generator low pressure and containment high pressure) are tested under Section 3.3, “Instrumentation.” The MFRV and MFRV bypass valves closure times are bounding values assumed in the MSLB containment response and core response (DNB) analyses (Refs. 3 and 4). This SR is normally performed during a refueling outage. The MFRVs and MFRV bypass valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not stroke tested at power, they are exempt from the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

INSERT Bases 3.7.3

BASES

REFERENCES

1. FSAR, Section 10.2.3

2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400

3. FSAR, Section 14.18.2

4. FSAR, Section 14.14

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a method for removing decay heat, should the preferred heat sink via the turbine bypass valve to the condenser not be available, as discussed in the FSAR, Section 10.2 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the Condensate Storage Tank (CST). The ADVs may also be used during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the turbine bypass valve.

Four ADVs are provided, two per steam generator. One ADV per steam generator is required to lower steam generator pressure to 885 psig in the event Auxiliary Feedwater Pump P-8C is needed to supply the steam generators for decay heat removal.

The ADVs are provided with upstream manual isolation valves to provide a means of isolation in the event an ADV spuriously opens, or fails to close during use. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are provided with a pressurized gas supply from the Bulk Nitrogen System that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ADVs. The nitrogen backup is not required for ADV OPERABILITY. A description of the nitrogen backup is found in the FSAR, Section 9.5.3 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is to prevent lifting of the Main Steam Safety Valves (MSSVs) following a turbine and reactor trip and to provide the capability to cool the plant to SDC System entry conditions when condenser vacuum is lost, making the turbine bypass valve unavailable.

BASES

APPLICABLE SAFETY ANALYSES (continued)	In certain accident analyses presented in the FSAR, the ADVs are assumed to be used by the operator for decay heat removal. The ADVs are credited in the loss of normal feed flow analysis when AFW pump P-8C is used and offsite power is available. Operator action may be required to either trip the four Primary Coolant Pumps (PCPs), start an additional AFW pump, or reduce steam generator pressure. This will allow the required AFW flowrate to the steam generators assumed by the loss of normal feed flow analysis. The ADVs are equipped with manual isolation valves in the event an ADV spuriously opens, or fails to close during use. The ADVs satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	One ADV is required to be OPERABLE on each steam generator to ensure that at least one ADV is OPERABLE to lower steam generator pressure to 885 psig following an event in which only AFW pump P-8C is available to supply the steam generators. A closed manual isolation valve renders its ADV inoperable, since operator action time to open the manual isolation valve is not supported in the accident analysis. Failure to meet the LCO can result in the inability to supply the required AFW flow rate to the steam generators assumed by the loss of normal feed flow analysis. An ADV is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and is capable of fully opening and closing on demand from either the control room or Hot Shutdown Panel (C-33).
APPLICABILITY	In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the ADVs are required to be OPERABLE. In MODES 5 and 6, there are no credible transients requiring ADVs.

BASES

ACTIONS

A.1

With one required ADV inoperable, action must be taken to restore the ADV to OPERABLE status within 7 days. The 7 day Completion Time takes into account the redundant capability afforded by the remaining OPERABLE ADV, and a nonsafety grade backup in the turbine bypass valve and MSSVs.

B.1

With two required ADVs inoperable, action must be taken to restore one of the ADVs to OPERABLE status. As the manual isolation valve can be closed to isolate an ADV, some repairs may be possible with the plant at power. The 24 hour Completion Time is reasonable to repair inoperable ADVs, based on the availability of the turbine bypass valve and MSSVs, and the low probability of an event occurring during this period that requires the ADVs.

C.1 and C.2

If the ADVs 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 upon the 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 3.7.4.1

To perform a controlled cooldown of the PCS, the ADVs must be able to be cycled through their full range. This SR ensures the ADVs are tested through a full control cycle. Performance of inservice testing or use of an ADV during a plant cooldown may satisfy this requirement. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 10.2
2. FSAR, Section 9.5.3

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Primary Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through a common suction line from the Condensate Storage Tank (CST) (LCO 3.7.6, "Condensate Storage and Supply") and pump to the steam generator secondary side via two separate and independent flow paths to a common AFW supply header for each steam generator. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the Main Steam Safety Valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or Atmospheric Dump Valves (ADVs) (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). If the main condenser is available, steam may be released via the turbine bypass valve.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into two trains. One train (A/B) consists of a motor driven pump (P-8A) and the turbine driven pump (P-8B) in parallel, the discharges join together to form a common discharge. The A/B train common discharge separates to form two flow paths, which feed each steam generator via each steam generator’s AFW penetration. The second motor driven pump (P-8C) feeds both steam generators through separate flow paths via each steam generator AFW penetration and forms the other train (C). The two trains join together at each AFW penetration to form a common supply to the steam generators. Each AFW pump is capable of providing 100% of the required capacity to the steam generators as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system.

Each motor driven AFW pump is powered from an independent Class 1E power supply, and feeds both steam generators.

BASES

BACKGROUND (continued)

The steam turbine driven AFW pump receives steam from the steam generator E-50A main steam header upstream of the Main Steam Isolation Valve (MSIV). The steam supply valve receives an open signal from the Auxiliary Feedwater Actuation Signal (AFAS) instrumentation. The turbine driven AFW pump feeds both steam generators through the same flow paths as motor driven AFW pump P-8A.

One pump at full flow is sufficient to remove decay heat and cool the plant to Shutdown Cooling (SDC) System entry conditions.

The AFW System supplies feedwater to the steam generators during normal plant startup, shutdown, and hot standby conditions.

The AFW System is designed to supply sufficient water to the steam generators to remove decay heat with steam generator pressure at the setpoint of the MSSVs, with exception of AFW pump P-8C. If AFW pump P-8C is used, operator action may be required to either trip the four Primary Coolant Pumps (PCPs), start an additional AFW pump, or reduce steam generator pressure. This will allow the required flowrates to the steam generators that are assumed in the safety analyses. Subsequently, the AFW System supplies sufficient water to cool the plant to SDC entry conditions, and steam is released through the ADVs, or the turbine bypass valve if the condenser is available.

The AFW System actuates automatically on low steam generator level by an AFAS as described in LCO 3.3.3, "Engineered Safety Feature (ESF) Instrumentation" and 3.3.4, "ESF Logic." The AFAS initiates signals for starting the AFW pumps and repositioning the valves to initiate AFW flow to the steam generators. The actual pump starts are on an "as required" basis. P-8A is started initially, if the pump fails to start, or if the required flow is not established in a specified period of time, P-8C is started. If P-8A and P-8C do not start, or if required flow is not established in a specified period of time, then P-8B is started.

The AFW System is discussed in the FSAR, Section 9.7 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat, by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest MSSV set pressure plus 3% with the exception of AFW pump P-8C. If AFW pump P-8C is used, operator action may be required to either trip the four PCPs, start an additional AFW pump or reduce steam generator pressure. This will allow the required flowrate to the steam generators that are assumed in the safety analyses.

The limiting Design Basis Accident for the AFW System is a loss of normal feedwater.

In addition, the minimum available AFW flow and system characteristics impact the analysis of a small break loss of coolant accident.

The AFW System design is such that it can perform its function following loss of normal feedwater combined with a loss of offsite power with one AFW pump injecting AFW to one steam generator.

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

LCO

This LCO requires that two AFW trains be OPERABLE to ensure that the AFW System will perform the design safety function to mitigate the consequences of accidents that could result in overpressurization of the primary coolant pressure boundary. Three independent AFW pumps, in two diverse trains, ensure availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two pumps from independent emergency buses. The third AFW pump is powered by a diverse means, a steam driven turbine supplied with steam from a source not isolated by the closure of the MSIVs.

BASES

LCO (continued)

The AFW System is considered to be OPERABLE when the components and flow paths required to provide AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to both steam generators. Prior to making the reactor critical during a plant startup, the turbine driven AFW pump shall be OPERABLE and capable of supplying AFW flow to both steam generators. When steam generator pressure is reduced, it is not required to have design inlet pressure available to the turbine driver in order to declare the turbine driven AFW pump OPERABLE. As steam generator pressure drops, the required AFW pump discharge head decreases accordingly. The reduced steam generator pressure available at lower temperatures in MODE 3 does not inhibit the turbine driven AFW pump's ability to feed the steam generator (Ref. 3). The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE.

The LCO is modified by three Notes. Note one indicates that only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of reduced heat removal requirements, the short period of time in MODE 4 during which AFW is required, and the insufficient steam pressure available in MODE 4 to power the turbine driven AFW pump. Note two states that the turbine driven AFW pump is only required to be made OPERABLE prior to making the reactor critical. It is required to be OPERABLE during subsequent MODE 1, 2, and 3 operation. This allowance is needed to provide sufficient steam pressure to perform turbine and pump testing. Note three indicates that any two AFW pumps may be placed in manual mode for the purpose of testing, for not more than 4 hours. In this situation, the third AFW pump would still be available in the event of a plant transient. The two pumps that are in manual could be used at the discretion of the operator.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE and to function in the event that the main feedwater is lost. In addition, the AFW System is required to supply enough makeup water to replace steam generator secondary inventory, lost as the plant cools to MODE 4 conditions.

During heatup, the turbine driven AFW pump is only required to be made OPERABLE prior to making the reactor critical. It is required to be OPERABLE during subsequent MODE 1, 2, and 3 operation. This allowance is needed to provide sufficient steam pressure to perform turbine and pump testing.

BASES

APPLICABILITY (continued)

In MODE 4, the AFW System may be used for heat removal via the steam generator.

In MODES 5 and 6, the steam generators are not normally used for decay heat removal, and the AFW System is not required.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

Condition A is applicable whenever one or more AFW trains is inoperable, in MODE 1, 2, or 3. Action A.1 requires restoration of both trains to OPERABLE status within 72 hours. The 72 hour Completion Time is based on the assumption that at least 100% of the required AFW flow (that assumed in the safety analyses) is available to each steam generator. If the flow available to either steam generator is less than 100% of the required AFW flow, or if less than two AFW pumps are OPERABLE, Condition B must also be entered. In addition, if the combined flow available to both steam generators is less than 100% of the required AFW flow, Condition C must be entered as well.

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

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

ACTIONS
(continued)

B.1 and B.2

Condition B is applicable: 1) when the Required Actions of Condition A cannot be completed within the required Completion Time, 2) when the flow available to either steam generator is less than 100% of the required AFW flow, or 3) when less than two AFW pumps are OPERABLE. Condition A is applicable whenever one or more trains is inoperable. Therefore, when Condition B is applicable, Condition A is also applicable. (If the combined flow available to both steam generators is less than 100% of the required AFW flow, Condition C must be entered as well.) Being in Conditions A and B concurrently maintains both Completion Time clocks for instances where equipment repair allows exit from Condition B while the plant is still within the applicable conditions of the LCO.

Continued plant operation is not allowed if the available AFW flow to either steam generator is less than the required flow, because adequate AFW flow cannot be assured following a main steam line break affecting that steam generator (consider the case where the break occurs in the AFW piping). Therefore, if 1) the inoperable AFW trains cannot be restored to OPERABLE status within the required Completion Time of Condition A, or 2) the flow available to either steam generator is less than 100% of the required AFW flow, or 3) less than two AFW pumps are OPERABLE in MODES 1, 2, and 3, the plant must be placed in a MODE in which the LCO does not apply (except as noted in Condition C). To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

C.1

Condition C is applicable if the combined flow available to both steam generators is less than 100% of the required AFW flow; Condition A is applicable whenever one or more trains is inoperable; and Condition B is applicable when the flow available to either steam generator is less than 100% of the required AFW flow, or when less than two AFW pumps are OPERABLE. Therefore, when Condition C is applicable, Conditions A and B are also applicable. Being in Conditions A, B, and C concurrently maintains the Completion Time clocks for instances where equipment repair allows exit from Condition C while the plant is still within the applicable conditions of the LCO.

One hundred percent AFW flow (that assumed in the safety analyses) can be provided by any one OPERABLE AFW pump and an OPERABLE flow path to each steam generator.

Required Action C.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until at least 100% of the required AFW flow is available. In this condition, there may be inadequate AFW flow available to remove decay heat and allow a stable plant shutdown.

With less than 100% of the required AFW flow available (ie. less than the AFW flow assumed in the safety analyses, while in MODES 1, 2, and 3, or less than the required AFW train OPERABLE while in MODE 4 with a steam generator relied upon for heat removal), the plant is in a seriously degraded Condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the plant should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least 100% of the required AFW flow available. LCO 3.0.3 is not applicable, as it could force the plant into a less safe condition.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for the required manual, power operated, and automatic valves in the AFW water and steam supply flow path provides assurance that the proper flow paths exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulations; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

This test need not be performed for the steam driven AFW pump for MODE 4 operation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.5.2

Verifying that each required AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by the ASME Code (Ref. 2). This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance.

This SR is modified by a Note indicating that this SR for the turbine driven AFW pump does not have to be met in MODE 3 when steam pressure is below 800 psig. This is because there is insufficient steam pressure and pump discharge pressure to allow the turbine driven pump to reach the normal test conditions.

Performance of inservice testing as discussed in the ASME Code (Ref. 2), and the INSERVICE TESTING PROGRAM satisfies this requirement.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an AFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. Specific signals (e.g., AFAS) are tested under Section 3.3, "Instrumentation." This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note which states the SR is only required to be met in MODES 1, 2, and 3 when AFW is not in operation. With AFW in operation, the required trains are already aligned with the flow control valves in manual control.

SR 3.7.5.4

This SR ensures that the AFW pumps will start in the event of any accident or transient that generates an AFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. Specific signals (e.g., AFAS, handswitch) are tested under Section 3.3, "Instrumentation."

This test need not be performed for the steam driven AFW pump for MODE 4 operation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The Note states that the SR is only required to be met in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required.

REFERENCES

1. FSAR, Section 9.7

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

3. Palisades Design Basis Document 1.03, Auxiliary Feedwater System, Section 3.4.1.

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 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. In a loss of offsite power, only the gravity flow path would be available. The inventory analysis in Reference 2 credits the CST and T-81 control valves, and their bypass valves, in their open positions for the gravity flow path between the tanks.

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 9.7
2. Analysis EA-GOTHIC-CST-01

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System (SWS), and thus to the environment.

The isolation of the CCW to components or systems may render those components inoperable but does not affect the OPERABILITY of the CCW System.

The CCW System consists of three pumps connected in parallel to common suction and discharge headers. The discharge header splits into two parallel heat exchangers and then combines again into a common distribution header which supplies various heat loads. A common surge tank provides the necessary net positive suction head for the CCW pumps and a surge volume for the system. A train of CCW is considered to be that equipment electrically connected to a common safety bus necessary to transfer heat acquired from the various heat loads to the SWS. There are two CCW trains, each associated with a Safeguards Electrical Distribution Train which are described in Specification 3.8.9, "Distribution Systems - Operating."

1. The CCW train associated with the Left Safeguards Electrical Distribution Train consists of one CCW pump (P-52A), CCW heat exchanger E-54B, the CCW surge tank (T-3), associated piping, CCW control valves receiving an actuation signal from the left train (eg. CV-0911, CV-0938, & CV-0946), and controls for that equipment to perform their safety function.

2. The CCW train associated with the Right Safeguards Electrical Distribution Train consists of one CCW pump (P-52B), CCW heat exchanger E-54A, the CCW surge tank (T-3), associated piping, CCW control valves receiving an actuation signal from the right train (eg. CV-0937, CV-0940, & CV-0945), and controls for that equipment to perform their safety function.

BASES

BACKGROUND (continued)

3. The CCW system piping, CCW surge tank (T-3), CCW control valves which receive actuation signals from both right and left trains (eg. CV-0910, CV-0913, CV-0944, CV-0944A, CV-0950, & CV-0977B), and controls for that equipment to perform their safety function.

CCW system components receive three automatic actuation signals, a Safety Injection Signal (SIS), a Recirculation Actuation Signal (RAS), or a Containment High Pressure (CHP) signal:

1. SIS starts the CCW pumps, isolates non-essential CCW loads outside the containment, opens the CCW inlet valves to the Shutdown Heat Exchangers (SDHXs), and sends an open signal to the engineered safeguards pump cooler CCW inlet valves (which are normally open).
2. RAS sends an open signal to the CCW heat exchanger CCW inlet valves (which are normally open).
3. CHP isolates the CCW loads inside the containment.

The CCW System cools three groups of loads which are described in the FSAR (Ref. 1). The major loads are:

1. Safety related loads outside the containment,
   - Shutdown Cooling Heat Exchangers
   - Engineered Safeguards Pump Coolers
2. Non-safety related loads outside the Containment, and
   - Spent Fuel Cooling Heat Exchangers
   - Waste Gas Compressors
   - Rad Waste Evaporators
   - Charging Pump Oil Coolers
3. Non-safety related loads inside the Containment.
   - Letdown Heat Exchanger
   - Shield Cooling Heat Exchangers
   - Primary Coolant Pump Leakoff and Oil Coolers
   - CRDM Seal Coolers

Each of these groups of loads can be cooled by the flow from one CCW pump. During normal operation, when full flow is not being provided to the Shutdown Cooling and Letdown Heat Exchangers, one CCW pump can provide the required flow for all three groups of loads. Two pumps may be operated to provide additional system flow and thermal stability.

BASES

BACKGROUND (continued)

During post accident conditions, with all CCW and related system components OPERABLE, one hundred percent of the required CCW post accident cooling capability can be provided by any one CCW pump with sufficient flow margin to allow manually restoring CCW flow to the Spent Fuel Pool Cooling Heat Exchangers. If CCW or related systems have components out of service, additional CCW pumps may be required to provide the required post accident cooling capability.

For post accident cooling, the Engineered Safety Features signals reposition several valves to maximize containment cooling and conserve CCW flow. Initially, a safety injection signal will start the CCW pumps, and open the large CCW inlet valves to the Shutdown Cooling Heat Exchangers (CCW cools the Shutdown Cooling Heat Exchangers, which cool the containment spray flow). A safety injection signal will also isolate the non-safety related CCW loads outside the containment. A Containment High Pressure signal will isolate the non-safety related CCW loads inside the containment. The occurrence of these automatic actions will provide the required CCW post accident cooling capability while limiting the CCW flow requirement to that which can be provided by one CCW pump.

The safety analyses assume that both CCW heat exchangers are available. To assure that both heat exchangers will be available even with a single active failure, the CCW inlet valves to the CCW heat exchangers are maintained in the full open position during plant operation.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section 9.3 (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Shutdown Cooling (SDC) System heat exchangers. This may utilize the SDC heat exchangers during a normal or post accident cooldown and shutdown in conjunction with the Containment Spray System during the recirculation phase following a LOCA.

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one CCW train in conjunction with the SWS and a 100% capacity Containment Cooling System (containment spray, containment coolers, or a combination) removing core decay heat between 20 to 40 minutes after a design basis LOCA. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Primary Coolant System (PCS) by the safety injection pumps. Any single CCW pump can provide one hundred percent of the required CCW post accident cooling capability if both CCW heat exchangers are available.

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power. The CCW System also functions to cool the plant from SDC entry conditions (T_{ave} < 300°F) to MODE 5 (T_{ave} < 200°F) during normal and post accident operations. The time required to cool from 300°F to 200°F is a function of the number of CCW and SDC trains operating. This assumes that the maximum Lake Michigan water temperature of LCO 3.7.9, “Ultimate Heat Sink (UHS),” occurs simultaneously with the maximum heat loads on the system.

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

BASES

LCO

The CCW trains are independent of each other to the degree that each has separate controls and power supplies. In the event of a DBA, one CCW train is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two CCW trains must be OPERABLE. At least one CCW train will operate assuming the worst single active failure occurs coincident with the loss of offsite power.

The CCW train associated with the Left Safeguards Electrical Distribution Train is considered OPERABLE when:

a. CCW pump P-52A is OPERABLE;

b. CCW surge tank T-3 and other common components are OPERABLE;

c. CCW heat exchanger E-54B is OPERABLE; and

d. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

The CCW train associated with the Right Safeguards Electrical Distribution Train is considered OPERABLE when:

a. CCW pump P-52B is OPERABLE;

b. CCW surge tank T-3 and other common components are OPERABLE;

c. CCW heat exchanger E-54A is OPERABLE; and

d. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable, but does not affect the OPERABILITY of the CCW System.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system that must be prepared to perform its post accident safety functions, primarily PCS heat removal by cooling the SDC heat exchanger.

In MODES 5 and 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

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

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

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

B.1 and B.2

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

BASES

ACTIONS

B.1 & B.2 (continued)

If the required CCW trains 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 5 within 36 hours.

C.1

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

If the CCW side (shell side) of either CCW heat exchanger is out of service, 100% of the required CCW post accident cooling capability cannot be assured. If the SWS side (tube side) of either CCW heat exchanger is out of service, 100% of the required CCW post accident cooling capability can be provided, if other equipment outages are limited. One hundred percent of the post accident CCW cooling can be provided with the SWS side of one CCW heat exchanger out of service if the following equipment is OPERABLE: 3 safety related Containment Air Coolers, 2 Containment Spray Pumps, CCW pumps P-52A and P-52B, 2 SWS pumps, and both Shutdown Cooling Heat Exchangers, and if

1. One CCW Containment header valve, CV-0910, CV-0911, or CV-0940, is OPERABLE, and

2. Two CCW isolation valves for the non-safety related loads outside the containment, CV-0944A and CV-0944 (or CV-0977B), are OPERABLE.

One hundred percent of the required CCW post accident cooling can be provided despite the inoperability of one or more of those CCW valves closed by Safety Injection, which isolate cooling to non-essential loads, provided there are sufficient CCW pumps available to supply the additional flow.

BASES

ACTIONS

C.1 (continued)

One hundred percent of the required CCW post accident cooling capability can be provided by one CCW pump if both CCW heat exchangers are available and if:

1. One CCW Containment header valve, CV-0910, CV-0911, or CV-0940, is OPERABLE, and

2. Two CCW isolation valves for the non-safety related loads outside the containment, CV-0944A and CV-0944 (or CV-0977B), are OPERABLE.

One hundred percent of the required CCW post accident cooling capability can be provided by two CCW pumps if both CCW heat exchangers are available and if:

1. One CCW Containment header valve, CV-0910, CV-0911, or CV-0940, is OPERABLE, or

2. Two CCW isolation valves for the non-safety related loads outside the containment, CV-0944A and CV-0944 (or CV-0977B), are OPERABLE.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

This SR is modified by a Note indicating that the isolation of the CCW to components or systems may render those components inoperable but does not affect the OPERABILITY of the CCW System.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. Specific signals (e.g., safety injection, RAS) are tested under Section 3.3, "Instrumentation." This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This SR is modified by a Note which states this SR is only required to be met in MODES 1, 2, and 3. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal in the "with standby power available" mode which tests the starting of the pumps by the SIS-X relays. The starting of the pumps by the sequencer is performed in Section 3.8, "Electrical Power Systems." This SR is modified by a Note which states this SR is only required to be met in MODES 1, 2, and 3. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 9.3

B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation or a normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The isolation of the SWS to components or systems may render those components inoperable but does not affect the OPERABILITY of the SWS System.

The SWS consists of three pumps connected in parallel taking suction from a common intake structure supplied by Lake Michigan. The discharge of the pumps flow into a common header before splitting into three headers (two critical headers for safety-related equipment and a single non-critical header for non safety-related equipment). The return piping from the three headers join into a common line and discharge to the cooling tower makeup basin. A train of SWS shall be that equipment electrically connected to a common safety bus necessary to remove heat from the various heat loads. There are two SWS trains, each associated with a Safeguards Electrical Train which are described in Specification 3.8.9, “Distribution Systems - Operating.” The SWS train associated with the Left Safeguards Train consists of one SWS pump (P-7B), associated piping, valves, and controls for the equipment to perform their safety function. The SWS train associated with the Right Safeguards Train consists of two SWS pumps (P-7A, P-7C), associated piping, valves, and controls for the equipment to perform their safety function. The pumps and valves are remote manually aligned, except in the unlikely event of a Loss Of Coolant Accident (LOCA).

SWS components receive three automatic actuation signals, a Safety Injection Signal (SIS), a Recirculation Actuation Signal (RAS), or a Diesel Generator (DG) start signal:

1. SIS starts the SWS pumps, isolates the non-critical service water header, and realigns the Containment Air Cooler (CAC) service water valves to the post accident cooling configuration.

BASES

BACKGROUND (continued)

2. RAS realigns the CCW heat exchanger service water outlet valves for maximum cooling.

3. A DG start signal opens the DG lube oil and jacket water cooler inlet valves.

The DG which powers two SWS pumps (P-7A, P-7C), also powers the fans associated with VHX-1, VHX-2, and VHX-3 (V-1A, V-2A and V-3A). This is necessary because if reliance for containment cooling is placed on CACs, at least two service water pumps must be OPERABLE to provide the necessary service water flow to assure OPERABILITY of the CACs. The Service Water System cools three groups of loads. The SWS loads are described in the FSAR (Ref. 1), the major loads are:

1. Critical loads inside the Containment,
   Containment Air Coolers VHX-1, VHX-2, VHX-3, (and VHX-4)

2. Critical loads outside the Containment, and
   Diesel Generators 1-1 and 1-2
   Component Cooling Heat Exchangers E-54A and E-54B
   Engineered Safeguards Room Coolers VHX-27A and VHX-27B
   Control Room HVAC Coolers VC-10 and VC 11
   Instrument Air Compressor C-2A and C-2C After Coolers

3. Non-critical loads in the Turbine Building

Each of these groups of loads can be cooled by the flow from one SWS pump. During normal operation, when SWS flow from the CACs and CCW heat exchangers is throttled by temperature control valves, two SWS pumps can provide the required flow for all three groups of loads.

During post accident conditions, with all other SWS and related system components OPERABLE, one hundred percent of the required SWS post accident cooling capability can be provided by any one SWS pump. If SWS or related systems have components out of service, additional SWS pumps may be required to provide the required cooling capability.

For post accident cooling, the Engineered Safety Features signals reposition several valves to maximize containment cooling and conserve SWS flow. Initially, a safety injection signal will start the SWS pumps, realign the SWS valves for the CACs (which cool the containment atmosphere), and close the non-critical SWS header isolation valve.

BASES

BACKGROUND (continued)

Subsequently, if the Safety Injection Refueling Water Tank has been emptied, a RAS will realign the SWS outlet valves on the CCW heat exchangers (CCW cools the Shutdown Cooling Heat Exchangers, which cool the containment spray flow). The occurrence of these automatic actions will provide the one hundred percent of the required post accident SWS cooling capability while limiting the SWS flow requirement to that which can be provided by two SWS pumps.

If the Containment Air Coolers are not needed for post accident containment cooling. SWS flow to the containment may then be isolated, further reducing the required SWS post accident cooling capability to that which can be provided by one SWS pump.

One hundred percent of the required SWS post accident cooling capability can be provided by any one SWS pump if SWS flow both to the non-critical header and to the critical loads inside the containment are capable of being isolated.

1. The capability to isolate SWS flow to the non-critical SWS header requires its isolation valve, CV-1359, to be OPERABLE.

2. The allowance to isolate SWS flow to the containment requires the ability to provide post accident containment cooling without reliance on CACs.

   The capability to isolate SWS flow to the containment requires one SWS Containment Isolation Valve, CV-0824 or CV-0847, to be OPERABLE.

One hundred percent of the required SWS post accident cooling capability can be provided by any two SWS pumps if SWS flow either to the non-critical header or to the critical loads inside the containment are capable of being isolated.

One hundred percent of the required SWS post accident cooling capability can be provided by three SWS pumps even with SWS flow being provided to both the CACs and the Non-critical SWS header.

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the FSAR, Section 9.1 (Ref. 1). The principal safety related functions of the SWS is the removal of decay heat from the reactor via the Component Cooling Water (CCW) System and the removal of heat from the containment atmosphere via the CACs.

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the SWS is for one SWS train, in conjunction with the CCW System and a 100% capacity containment cooling system (containment spray, CACs, or a combination), removing core decay heat between 20 to 40 minutes following a design basis LOCA. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Primary Coolant System by the safety injection pumps. The SWS is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The SWS, in conjunction with the CCW System, also cools the plant from Shutdown Cooling (SDC) entry Condition, as discussed in the FSAR, Section 6.1 (Ref. 2) to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CCW and SDC System trains that are operating. This assumes that the maximum Lake Michigan water temperature of LCO 3.7.9, “Ultimate Heat Sink (UHS),” occurs simultaneously with maximum heat loads on the system.

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

LCO

Two SWS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power.

The SWS train associated with the Left Safeguard Electrical Distribution Train is considered OPERABLE when:

a. SWS pump P-7B is OPERABLE; and
b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

The SWS train associated with the Right Safeguards Electrical Distribution Train is OPERABLE when:

a. SWS pumps P-7A and P-7C are OPERABLE; and
b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of SWS from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the SWS System.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS System is a normally operating system, which is required to support the OPERABILITY of the equipment serviced by the SWS and required to be OPERABLE in these MODES. In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports.

ACTIONS

A.1

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

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

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

B.1 and B.2

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

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable SWS trains cannot be restored to OPERABLE status within the associated required Completion Time of Condition A, 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 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.

C.1

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

The Service Water System cools three groups of loads:

1. Critical loads inside the Containment,
2. Critical loads outside the Containment, and
3. Non-critical loads in the Turbine Building.

As discussed in the Background section of these bases, each of these groups of loads can be cooled by the flow from one SWS pump.

One hundred percent of the required SWS post accident cooling capability can be provided by any one SWS pump if:

1. The non-critical SWS header isolation valve, CV-1359, is OPERABLE, and

2. Plant conditions allow adequate containment cooling to be provided without reliance on CACs and one SWS Containment Isolation Valve, CV-0824 or CV-0847, is OPERABLE.

One hundred percent of the required SWS post accident cooling capability can be provided by any two SWS pumps if:

BASES

ACTIONS (continued)

C.1

1. The non-critical SWS header isolation valve, CV-1359, is OPERABLE, or

2. Plant conditions allow adequate containment cooling to be provided without reliance on CACs and one SWS Containment Isolation Valve, CV-0824 or CV-0847, is OPERABLE.

One hundred percent of the required SWS post accident cooling capability can be provided by three SWS pumps even with SWS flow being provided to both the CACs and the Non-critical SWS header.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path ensures that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of SWS to components or systems may render those components inoperable but does not affect the OPERABILITY of the SWS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

INSERT Bases 3.7.8

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. Specific signals (e.g., safety injection) are tested under Section 3.3, "Instrumentation." This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This SR is modified by a Note which states this SR is only required to be met in MODES 1, 2, and 3. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.8.3

The SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal in the "with standby power available" mode which tests the starting of the pumps by the SIS-X relays. The starting of the pumps by the sequencer is performed in Section 3.8, "Electrical Power Systems." This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 9.1

2. FSAR, Section 6.1

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND	The UHS provides a heat sink for process and operating heat from safety related components during a Design Basis Accident (DBA) or transient, as well as during normal operation. This is done utilizing the Service Water System (SWS). The UHS has been defined as Lake Michigan. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident. The basic performance requirements are that an adequate Net Positive Suction Head (NPSH) to the SWS pumps be available, and that the design basis temperatures of safety related equipment not be exceeded. Additional information on the design and operation of the system along with a list of components served can be found in FSAR, Section 9.1 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the plant is cooled down and placed on shutdown cooling. Maximum post accident heat load occurs between 20 to 40 minutes after a design basis Loss of Coolant Accident (LOCA). Near this time, the plant switches from injection to recirculation, and the containment cooling systems are required to remove the core decay heat.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The minimum water level of the UHS is based on the NPSH requirements for the SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening which is at elevation 568.25 ft. Early warning of a falling intake water level is provided by the intake structure level alarm. The nominal lake level is approximately 580 ft mean sea level. The maximum water temperature of the UHS is based on conservative heat transfer analyses for the worst case LOCA. FSAR, Section 14.18 (Ref. 2) and Design Basis Document (DBD) 1.02 (Ref. 3) provide the details of the analysis which forms the basis for these operating limits. The assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case single active failure. The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2).
LCO	The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate without the loss of NPSH, and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed 85°F and the level should not fall below 568.25 ft above mean sea level during normal plant operation.
APPLICABILITY	In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES. In MODES 5 and 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

BASES

ACTIONS

A.1 and A.2

If the UHS is inoperable, 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 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies adequate cooling can be maintained. The level specified also ensures sufficient NPSH is available for operating the SWS pumps. This SR verifies that the UHS water level is ≥ 568.25 ft above mean sea level as measured within the boundaries of the intake structure. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.9.2

This SR verifies that the SWS is available to provide adequate cooling for normal design heat loads and maximum accident conditions following a DBA. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. This SR verifies that the water temperature from the UHS is ≤ 85°F.

REFERENCES

1. FSAR, Section 9.1

2. FSAR, Section 14.18

3. Design Basis Document (DBD) 1.02, “Service Water System”

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Ventilation (CRV) Filtration

BASES

BACKGROUND

The CRV Filtration provides a protected environment from which occupants can control the plant following an uncontrolled release of radioactivity.

The CRV Filtration consists of a common emergency intake which splits into two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. The exhaust of each train exhausts into a common supply plenum. Each train consists of a prefilter, a heater, a High Efficiency Particulate Air (HEPA) filter, two banks of activated charcoal adsorbers for removal of gaseous activity (principally iodines), a second HEPA filter, and a fan. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines, and provides back up in case of failure of the main HEPA filter bank.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the analyses of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CRV Filtration is an emergency system, part of which may also operate during normal plant operations in the standby mode of operation. Upon manual initiation or receipt of a containment high pressure or containment high radiation signal, normal air supply to the CRE is isolated, and the stream of ventilation air is recirculated through the filter trains of the system. The prefilters remove any large particles in the air. Continuous operation of each train for at least 10 hours per month, with

BACKGROUND (continued)

the heaters on, reduces moisture buildup on the HEPA filters and adsorbers.

Actuation of the system to the emergency mode of operation closes the normal unfiltered outside air intake and unfiltered exhaust dampers, opens the emergency air intake, and aligns the system for recirculation of the air within the CRE through the redundant trains of HEPA and charcoal filters. The emergency mode also initiates pressurization and filtered ventilation of the air supply to the CRE.

Outside air is filtered, and then added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary.

A single train will pressurize the CRE to at nominally 0.125 inches water gauge relative to external areas adjacent to the CRE boundary, and provides an air exchange rate in excess of 25% per hour. The CRV Filtration operation in maintaining the CRE habitable is discussed in the FSAR, Section 9.8 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across one filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CRV Filtration is designed in accordance with Seismic Category I requirements.

The CRV Filtration is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem Total Effective Dose Equivalent (TEDE), which is consistent with 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The CRV Filtration components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access.

The CRV Filtration provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis events discussed in the FSAR, Chapter 14 (Ref. 2).

APPLICABLE SAFETY ANALYSES (continued)

The CRV system provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release. The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels. No CRV Filtration actuation is required for hazardous chemical releases or smoke.

The worst case single active failure of a component of the CRV Filtration, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

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

LCO

Two independent and redundant trains of the CRV Filtration are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem TEDE in the event of a large radioactive release.

Each CRV Filtration train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CRV Filtration train is considered OPERABLE when the associated:

a. Main recirculation fan and emergency filter fan are OPERABLE;

b. HEPA filters and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Required heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CRV Filtration trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

This LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative control. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design

BASES

LCO (continued)

condition, such as doors, hatches, floor plugs, and access panels. Since this Note modifies the LCO, no Condition entry is required when the control room boundary is opened under its provisions. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CRV Filtration must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA.

In MODES 5 and 6, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining CRV Filtration OPERABLE is not required in MODE 5 or 6, except for the following situations under which significant radioactive releases can be postulated:

a. During CORE ALTERATIONS;

b. During movement of irradiated fuel assemblies; and

c. During movement of a fuel cask in or over the SFP.

ACTIONS

A.1

With one CRV Filtration train inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CRV Filtration train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CRV Filtration train could result in loss of CRV Filtration function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

ACTIONS (continued)

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

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the analyses of DBA consequences (allowed to be up to 5 rem TEDE), the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological event. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

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

If both CRV Filtration trains are inoperable in MODE 1, 2, 3, or 4, for reasons other than an inoperable control room boundary (i.e. Condition B), at least one CRV Filtration train must be returned to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second CRV Filtration train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. During the period that the CRV Filtration

ACTIONS (continued)

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

trains are inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from potential hazards while both trains of CRV Filtration are inoperable. In the event of a DBA, the mitigating actions will reduce the consequences of radiological exposures to the CRE occupants.

Specification 3.4.16, “PCS Specific Activity,” allows limited operation with the primary coolant system (PCS) activity significantly greater than the LCO limit. This presents a risk to the plant operator during an accident when all CRV Filtration trains are inoperable. Therefore, it must be verified within 1 hour that LCO 3.4.16 is met. This Required Action does not require additional PCS sampling beyond that normally required by LCO 3.4.16.

At least one CRV Filtration train must be returned to OPERABLE status within 24 hours. The Completion Time is based on Reference 3 which demonstrated that the 24 hour Completion Time is acceptable based on the infrequent use of the Required Actions and the small incremental effect on plant risk.

D.1, D.2.1, D.2.2, and D.2.3

During CORE ALTERATIONS, during movement of irradiated fuel assemblies, during movement of a fuel cask in or over the SFP, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CRV Filtration train must be immediately placed in the emergency mode of operation. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the plant in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies or a fuel cask to a safe position.

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

During CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP, with two CRV Filtration trains inoperable or with one or more CRV Filtration

ACTIONS (continued)

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

trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the CRE. This places the plant in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies or a fuel cask to a safe position.

F.1 and F.2

If an inoperable CRV Filtration or the CRE boundary cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, 3, or 4, the plant must be placed in a MODE that minimizes the accident risk. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train periodically provides an adequate check on this system.

Heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Each train must be operated for ≥ 10 continuous hours with the associated heater, VHX-26A or VHX-26B, energized. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.2

This SR verifies that the required CRV Filtration testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRV Filtration filter tests are in accordance with the VFTP. The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

This SR verifies that each CRV Filtration train starts and operates on an actual or simulated actuation signal. Specific signals (e.g., containment high pressure, containment high radiation) are tested under Section 3.3, “Instrumentation.” This SR is modified by a Note which states this SR is only required to be met in MODES 1, 2, 3 and 4 and during movement of irradiated fuel assemblies in containment. The instrumentation providing the input signal is not required in other plant conditions, therefore, to keep consistency with Section 3.3, “Instrumentation,” the SR is not required to be met. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 4) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 5). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 6). Options for restoring the CRE boundary to OPERABLE status include changing the DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

BASES

REFERENCES

1. FSAR, Section 9.8

2. FSAR, Chapter 14

3. WCAP-16125-NP-A, “Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown,” Revision 2, August 2010.

4. Regulatory Guide 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors"

5. NEI 99-03, "Control Room Habitability Assessment," June 2001.

6. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040300694).

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Ventilation (CRV) Cooling System

BASES

BACKGROUND

The CRV Cooling provides temperature control for the control room during normal and emergency conditions.

The CRV Cooling consists of two independent, redundant trains, which exhaust into a common supply plenum that provide cooling and heating of recirculated control room air. In the emergency mode, the two trains are supplied by a common emergency intake which splits into the two trains. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for control room temperature control. The CRV Cooling is a subsystem providing air temperature control for the control room.

The CRV Cooling is an emergency system, parts of which may also operate during normal plant operations. A single train will provide the required temperature control to maintain the control room at 90°F or below. The CRV Cooling operation to maintain the control room temperature is discussed in the FSAR, Section 9.8 (Ref. 1).

The control room ventilation emergency mode of operation is actuated either by a containment high radiation signal or a containment high pressure signal, or manually from the control room. During emergency mode operation, the air handling units and the charcoal filter units of both Train A and Train B are actuated automatically. The CRV Cooling refrigerant Condensing Units VC-10 and VC-11 shut down and are manually restarted by the operator when their operation is required for control room cooling. In addition, since immediate operation of the CRV Cooling System is not necessary, other manual operations may be required to initiate control room cooling, depending on the configuration of the system upon initiation of the emergency mode signal.

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the CRV Cooling is to maintain temperature of the control room environment throughout 30 days of continuous occupancy.

The CRV Cooling components are arranged in redundant safety related trains. During normal and emergency operation, the CRV Cooling maintains the temperature at 90°F or below, as required by LCO 3.3.1, “Reactor Protection System (RPS) Instrumentation.” A single active failure of a component of the CRV Cooling, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The CRV Cooling is designed in accordance with Seismic Category I requirements. The CRV Cooling is capable of removing sensible and latent heat loads from the control room, considering equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

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

LCO

Two independent and redundant trains of the CRV Cooling are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident. In addition, since immediate operation of the CRV Cooling System is not necessary, other manual operations may be required to initiate control room cooling, depending on the configuration of the system upon initiation of the emergency mode signal.

The CRV Cooling is considered OPERABLE when the individual components that are necessary to maintain the control room temperature are OPERABLE in both trains. These components include the condensing units, fans, and associated temperature control instrumentation. In addition, the CRV Cooling must be OPERABLE to the extent that air circulation can be maintained.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the CRV Cooling must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY requirements following isolation of the control room.

In MODES 5 and 6, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining CRV Cooling OPERABLE is not required in MODE 5 or 6, except for the following situations under which significant radioactive releases can be postulated:

a. During CORE ALTERATIONS;

b. During movement of irradiated fuel assemblies; and

c. During movement of a fuel cask in or over the SFP.

ACTIONS

A.1

With one CRV Cooling train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRV Cooling train is adequate to maintain the control room temperature within limits. The 30 day Completion Time is reasonable, based on the low probability of an event occurring requiring control room isolation, consideration that the remaining train can provide the required capabilities.

B.1

If two CRV cooling trains are inoperable, at least one CRV cooling train must be returned to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second CRV cooling train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. The Completion Time is based on Reference 2 which demonstrated that the 24 hour Completion Time is acceptable based on the infrequent use of the Required Action and the small incremental effect on plant risk.

BASES

ACTIONS (continued)

C.1 and C.2

In MODE 1, 2, 3, or 4, when one or more CRV cooling trains cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE that minimizes the accident risk. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in 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.

D.1, D.2.1, D.2.2, and D.2.3

During CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CRV Cooling train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the plant in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies or a fuel cask to a safe position.

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

During CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP, with two CRV Cooling trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the plant in a Condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies or a fuel cask to a safe position.

INSERT Bases 3.7.11

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.11.1 This SR verifies that the heat removal capability of the system is sufficient to meet design requirements. This SR consists of a combination of testing and calculations. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 9.8

2. WCAP-16125-NP-A, "Justification for Risk-Informed Modification to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

B 3.7 PLANT SYSTEMS

B 3.7.12 Fuel Handling Area Ventilation System

BASES

BACKGROUND

The Fuel Handling Area Ventilation System filters airborne radioactive particulates from the area of the spent fuel pool following a fuel handling accident or a fuel cask drop accident. The fuel handling area is served by two separate subsystems one being part of the original plant design, and the other being added as part of the Auxiliary Building Addition.

The original plant design consists of a supply plenum and an exhaust plenum including associated ductwork, dampers, and instrumentation. The supply plenum contains one prefilter, two heating coils, and one supply fan. The exhaust plenum contains two filter banks (normal and emergency) configured in a parallel flow arrangement, and two independent exhaust fans which draw air from a common duct. The “normal filter bank” contains a prefilter and a High Efficiency Particulate Air (HEPA) filter. The “emergency filter bank” contains a prefilter, HEPA filter, and an activated charcoal filter.

The Auxiliary Building Addition, which was added to serve the spaces at the north end of the spent fuel pool, also consist of a supply plenum and exhaust plenum. The supply plenum is configured similar to the supply plenum provided in the original plant design and includes one prefilter, two heating coils, and one supply fan. The exhaust plenum is different from the original plant design in that it only contains one filter bank consisting of a prefilter and HEPA filter, and two common exhaust fans.

During normal plant operations, the Fuel Handling Area Ventilation System supplies filtered and heated (as needed) outside air to the fuel handling area. The exhaust fans draw air from the fuel handling area through the normally aligned prefilters and HEPA filters and discharge it to the unit stack by way of the main ventilation exhaust plenum.

BASES

BACKGROUND (continued)

During plant evolutions when the possibility for a fuel handling accident or fuel cask drop accident exist, the Fuel Handling Area Ventilation System is configured such that all fans are stopped except one exhaust fan in the original plant subsystem aligned to the “emergency filter bank.” The “normal filter bank” in the original plant design is isolated by closing its associated inlet damper. Thus, in the event of a fuel handling accident, the fuel handling area atmosphere will be filtered for the removal of airborne fission products prior to being discharged to the outside environment.

The Fuel Handling Area Ventilation System is discussed in the FSAR, Sections 9.8, 14.11 and 14.19 (Refs. 1, 2, and 3) because it may be used for normal, as well as post-accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The Fuel Handling Area Ventilation System is designed to mitigate the consequences of a fuel handling accident or fuel cask drop accident by limiting the amount of airborne radioactive material discharged to the outside atmosphere.

The results and major assumptions used in the analysis of the fuel handling accident are presented in FSAR Section 14.19. For the purpose of defining the upper limit of the radiological consequences of a fuel handling accident, it is assumed that a fuel bundle is dropped during fuel handling activities and all the fuel rods in the equivalent of an entire assembly (216) fail. The bounding fuel handling accident is assumed to occur in containment two days after shutdown. No containment isolation is assumed to occur. As such, the released fission products escape to the environment with no credit for filtration. The results of this analysis have shown that the offsite doses resulting from this event are within the applicable limits of 10 CFR 50.67. In the event a fuel handling accident were to occur in the fuel handling area, the radioactive release would pass through the “emergency filter bank” significantly reducing the amount of radioactive material released to the environment. Thus, the consequences of a fuel handling accident in the fuel handling area are deemed acceptable with or without the “emergency filter bank” in operation since they are no more severe than the consequences of a fuel handling accident in containment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The results and major assumptions used in the analysis of the fuel cask drop accident are presented in FSAR Section 14.11. For the purpose of defining the upper limit of the radiological consequences of a fuel cask drop accident, it is assumed that all 73 fuel assemblies in a 7 x 11 Westinghouse spent fuel pool rack with a minimum decay of 30 days are damaged and release their fuel rod gap inventories. Three fuel cask drop scenarios were analyzed to encompass all fuel cask drop events. They are:

1. A fuel cask drop onto 30 day decayed fuel with the Fuel Handling Area Ventilation System aligned for emergency filtration with a conservative amount of unfiltered leakage. All isolatable unfiltered leak path are assumed to be isolated prior to event initiation.

2. A fuel cask drop onto 30 day decayed fuel with the Fuel Handling Area Ventilation System aligned for emergency filtration with a conservative amount of unfiltered leakage. This scenario determined the maximum amount of non-isolatable unfiltered leakage that can exist and still meet offsite dose limits. This scenario also assumes isolation of isolable leak paths prior to event initiation.

3. A fuel cask drop onto 90 day decayed fuel without the Fuel Handling Area Ventilation System aligned for emergency filtration. This scenario needs no assumptions as to unfiltered leakage or post-accident unfiltered leak path isolation times since all radiation is assumed to be released unfiltered from the fuel handling area.

The results of the analysis show that the radiological consequences of a fuel cask drop in the spent fuel pool meet the acceptance criteria of Regulatory Guide 1.183 (Ref. 4) and the applicable limits of 10 CFR 50.67 (Ref. 5) for all scenarios.

APPLICABLE SAFETY ANALYSES (continued)

Filtration of the fuel handling area atmosphere following a fuel handling accident is not necessary to maintain the offsite doses within the applicable limits of 10 CFR 50.67. Thus, a total system failure would not impact the margin of safety as described in the safety analysis. However, analysis has shown that post-accident filtration by the Fuel Handling Area Ventilation System provides significant reduction in offsite doses by limiting the release of airborne radioactivity. Therefore, for the fuel handling accident, the Fuel Handling Area Ventilation System satisfies Criterion 4 of 10 CFR 50.36(c)(2).

Filtration of the fuel handling area atmosphere following a fuel cask drop on irradiated fuel assemblies with < 90 days decay is required to maintain the offsite doses within the applicable limits of 10 CFR 50.67. Therefore, for the fuel cask drop accident, the Fuel Handling Area Ventilation System satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The LCO for the Fuel Handling Area Ventilation System ensures filtration of the fuel handling area atmosphere is immediately available in the event of a fuel handling accident, or a fuel cask drop accident. As such, the LCO requires the Fuel Handling Area Ventilation System to be OPERABLE with one fuel handling area exhaust fan aligned to the “emergency filter bank” and in operation.

The Fuel Handling Area Ventilation System is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE. The Fuel Handling Area Ventilation System is considered OPERABLE when:

a. One exhaust fan is aligned to the “emergency filter bank” and in operation to ensure the air discharged to the main ventilation exhaust plenum has been filtered. Operation of only one fuel handling area exhaust fan ensures the design flow rate of the “emergency filter bank” is not exceeded.

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork and dampers are OPERABLE, and air circulation can be maintained. Inclusive to the requirement to align the “emergency filter bank” is that the “normal filter bank” is isolated by its associated inlet damper to prevent the release of unfiltered air.

BASES

APPLICABILITY

The Fuel Handling Area Ventilation System must be OPERABLE, aligned, and in operation whenever the potential exists for an accident that results in the release of radioactive material to the fuel handling area atmosphere that could exceed previously approved offsite dose limits if released unfiltered to the outside atmosphere. As such, the Fuel Handling Area Ventilation System is required; during movement of irradiated fuel assemblies in the fuel handling building when irradiated fuel assemblies with < 30 days decay time are in the fuel handling building; during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in containment when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open, and during movement of a fuel cask in or over the spent fuel pool when irradiated fuel assemblies with < 90 days decay time fuel handling building.

The requirement for the Fuel Handling Area Ventilation System does not apply during movement of irradiated fuel assemblies or CORE ALTERATIONS when all irradiated fuel assemblies in the fuel handling building, or all irradiated fuel assemblies in the containment with the equipment hatch open, have decayed for 30 days or greater since the dose consequences from a fuel handling accident would be of the same magnitude without the filters operating as the dose consequences would be with the filters operating and two days decay. In addition, the requirement for the Fuel Handling Area Ventilation System does not apply during fuel cask movement when all irradiated fuel assemblies in the fuel handling building have decayed 90 days or greater since the dose consequences remain less than the applicable limits of 10 CFR 50.67.

ACTIONS

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

If the Fuel Handling Area Ventilation System is not aligned to the "emergency filter bank", or one exhaust fan is not in operation, or the system is inoperable for any reason, action must be taken to place the unit in a condition in which the LCO does not apply. Therefore, activities involving the movement of irradiated fuel assemblies, CORE ALTERATIONS, and movement of a fuel cask in or over the spent fuel pool, must be suspended immediately to minimize the potential for a fuel handling accident.

The suspension of fuel movement, CORE ALTERATIONS, and fuel cask movement shall not preclude the completion of placing a fuel assembly, core component, or fuel cask in a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies the performance of Fuel Handling Area Ventilation System filter testing in accordance with the Ventilation Filter Testing Program. The Fuel Handling Area Ventilation System filter tests are in accordance with the Regulatory Guide 1.52 (Ref. 6) as described in Ventilation Filter Testing Program. The Ventilation Filter Testing Program includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the Ventilation Filter Testing Program.

SR 3.7.12.2

This SR verifies the Fuel Handling Area Ventilation System has not degraded and is operating as assumed in the safety analysis. The flow rate is periodically tested to verify proper function of the Fuel Handling Ventilation System. When aligned to the “emergency filter bank”, the Fuel Handling Area Ventilation System is designed to reduce the amount of unfiltered leakage from the fuel handling building which, in the event of a fuel handling accident, lowers the dose at the site boundary to within the applicable limits of 10 CFR 50.67. The Fuel Handling Area Ventilation System is designed to lower the dose to these levels at a flow rate of $\geq$ 5840 cfm and $\leq$ 8760 cfm. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 9.8
2. FSAR, Section 14.11
3. FSAR, Section 14.19
4. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors.
5. 10 CFR 50.67
6. Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.13 Engineered Safeguards Room Ventilation (ESRV) Dampers

BASES

BACKGROUND

The ESRV Dampers isolate the safeguards rooms by closing the inlet and exhaust plenum dampers on the initiation of a high radiation alarm from their respective airborne particulate monitor. This isolation contributes to a lower offsite dose within applicable 10 CFR 50.67 (Ref. 1) limits if a leak should occur. Typically, high radiation would only be expected due to excessive leakage during the recirculation phase of operation following a Loss of Coolant Accident (LOCA).

The ESRV Dampers consists of two trains. Each train consists of a supply plenum damper, a exhaust plenum damper, and associated piping, valves, and ductwork. Instrumentation which is addressed in LCO 3.3.10, "Engineered Safeguards Room Ventilation (ESRV) Instrumentation," also form part of the system, but is not addressed by this LCO. The Reactor Auxiliary Building Main Ventilation System provides normal cooling in conjunction with the engineered safeguards room coolers. Upon receipt of a high radiation signal, the ESRV Dampers are closed, isolating the affected safeguards room(s) from the rest of the auxiliary building ventilation system lowering the leakage to the environment from the auxiliary building.

The ESRV Dampers are discussed in the FSAR, Sections 7.4.5.2 and 14.22 (Refs. 2 and 3).

APPLICABLE SAFETY ANALYSES

The design basis of the ESRV Dampers is established by the Maximum Hypothetical Accident (MHA). The system evaluation assumes leakage into the engineered safeguards rooms, such as safety injection pump seal leakage, during the recirculation mode. In such a case, the system limits the radioactive release from the engineered safeguards rooms to within applicable 10 CFR 50.67 limits (Ref. 1). The analysis of the effects and consequences of a MHA is presented in Reference 3. The ESRV Dampers may also actuate following a small break LOCA, after the plant goes into the recirculation mode of long term cooling to mitigate releases of smaller leaks, such as from valve stem packing.

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

BASES

LCO

Two ESRV Damper trains are required to be OPERABLE to ensure that each engineered safeguards room isolates upon receipt of its respective high radiation alarm. Total system failure could result in the atmospheric release from the engineered safeguards rooms exceeding the required limits in the event of a Design Basis Accident (DBA).

An ESRV Damper train is considered OPERABLE when its associated instrumentation, ductwork, valves, and dampers are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESR-Damper trains are required to be OPERABLE consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).

In MODES 5 and 6, the ESRV Damper trains are not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

Condition A addresses the failure of one or both ESRV Damper trains. Operation may continue as long as action is immediately initiated to isolate the affected engineered safeguards room. With the inlet and exhaust dampers closed, or if the inlet and outlet ventilation plenums are adequately sealed, the engineered safeguards room is isolated and the intended safety function is achieved, since the potential pathway for radioactivity to escape to the environment from the engineered safeguards room has been minimized.

The Completion Time for this Required Action is commensurate with the importance of maintaining the engineered safeguards room atmosphere isolated from the outside environment when the ECCS pumps are circulating primary coolant after an accident.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that each ESRV Damper train closes on an actual or simulated actuation signal. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50.67

2. FSAR, Section 7.4.5.2

3. FSAR, Section 14.22

PLANT

B 3.7 ~~FACILITY~~ SYSTEMS

B 3.7.14 Spent Fuel Pool (SFP) Water Level

BASES

BACKGROUND

The minimum water level in the SFP meets the assumptions of iodine decontamination factors following a fuel handling or cask drop accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the SFP design is given in the FSAR, Section 9.11 (Ref. 1), and the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.4 (Ref. 2). The assumptions of fuel handling and fuel cask drop accidents are given in the FSAR, Section 14.19 and 14.11 (Refs. 3 and 4), respectively.

APPLICABLE SAFETY ANALYSES

The minimum water level in the SFP meets the assumptions of fuel handling or fuel cask drop accident analyses described in References 3 and 4 and are consistent with the assumptions of Regulatory Guide 1.183 (Ref. 5). The resultant doses are within applicable 10 CFR 50.67 (Ref. 6) limits.

Reference 5 considers 23 ft of water between the top of the damaged fuel assembly and the fuel pool surface for a fuel handling or fuel cask drop accident. This LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single assembly, dropped and lying horizontally on top of the spent fuel racks, there may be < 23 ft of water above the top of the assembly and the surface, by the width of the assembly. Reference 5 specifies methods to address this condition. For additional conservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rods fail from a hypothetical maximum drop.

The SFP water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).

or for a fuel handling accident inside containment,

BASES

LCO

The specified water level preserves the assumptions of the fuel handling or fuel cask drop accident analyses. As such, it is the minimum required for movement of fuel assemblies or movement of a fuel cask in or over the SFP.

The LCO is modified by a Note which allows SFP level to be below the 647 ft elevation to support movement of a fuel cask in or over the SFP. This is necessary due to the water displaced by the fuel cask as it is lowered or dropped into the SFP. If the SFP level is normal prior to the fuel cask entering the SFP, the SFP could overflow.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the SFP or movement of a fuel cask in or over the SFP since the potential for a release of fission products exists.

ACTIONS

A.1 and A.2

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the SFP water level is lower than the required level, the movement of irradiated fuel assemblies in the SFP or movement of a fuel cask in or over the SFP are immediately suspended. This effectively precludes a spent fuel handling or fuel cask drop accident from occurring. This does not preclude moving a fuel assembly or fuel cask to a safe position.

The Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies or fuel cask in or over the SFP while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies or fuel cask in or over the SFP while in MODES 1, 2, 3, and 4, the movement of fuel or movement of a fuel cask is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies or fuel cask in or over the SFP is not sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient SFP water is available in the event of a fuel handling or fuel cask drop accident. The water level in the SFP must be checked periodically. ~~The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable, based on operating experience.~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

During refueling operations, the level in the SFP is at equilibrium with that of the refueling cavity, and the level in the refueling cavity is checked periodically in accordance with LCO 3.9.6, "Refueling Cavity Water Level."

REFERENCES

1. FSAR, Section 9.11
2. FSAR, Section 9.4
3. FSAR, Section 14.19
4. FSAR, Section 14.11
5. Regulatory Guide 1.183
6. 10 CFR 50.67

PLANT

B 3.7 ~~FACILITY~~ SYSTEMS

B 3.7.15 Spent Fuel Pool (SFP) Boron Concentration

BASES

BACKGROUND	As described in LCO 3.7.16, "Spent Fuel Pool Storage," fuel assemblies are stored in the fuel storage racks in accordance with criteria based on initial enrichment, discharge burnup, and decay time. The criteria were based on the assumption that 850 ppm of soluble boron was present in the spent fuel pool. The pool is required to be maintained at a boron concentration of ≥ 1720 ppm. Criterion 2 of 10 CFR 50.36 (c) (2) requires that criticality control be achieved without credit for soluble boron. However, in 1998 the NRC documented requirements that could be established to maintain criticality below 0.95. This is documented in "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", Laurence I. Kopp, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Reactor Systems Branch, February 1998. The precedent of taking credit for soluble boron in spent fuel pool water to provide criticality control has also been established. Soluble boron credit was used in the Westinghouse Spent Fuel Rack Criticality Analysis Methodology described in WCAP-14416-NP-A and that methodology was approved for use by an NRC Safety Evaluation dated October 25, 1996. The criteria discussed above was developed using a method that closely followed the Westinghouse methodology. Additionally the requirements specified by the NRC guidance are in place at Palisades.
APPLICABLE SAFETY ANALYSES	A fuel assembly could be inadvertently loaded into a fuel storage rack location not allowed by LCO 3.7.16 (e.g., an insufficiently depleted or insufficiently decayed fuel assembly). Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The concentration of dissolved boron in the SFP satisfies Criterion 2 of 10 CFR 50.36(c)(2).
LCO	The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the SFP.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

BASES

ACTIONS

A.1. and A.2.

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit.

The Surveillance Frequency is controlled by the Surveillance Frequency Control Porgram

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. ~~The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.~~

REFERENCES

None

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

B 3.7 ~~FACILITY~~ SYSTEMS

B 3.7.16 Spent Fuel Pool Storage

BASES

BACKGROUND

The fuel storage facility is designed to store used (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 892 fuel assemblies, which includes storage for failed fuel canisters. The fuel storage racks are grouped into two regions, Region I and Region II per Figure B 3.7.16-1. The racks are designed as a Seismic Category I structure able to withstand seismic events.

Region I contains Metamic equipped racks in the spent fuel pool having a 10.25 inch center-to-center spacing and a single Carborundum equipped rack in the north tilt pit having an 11.25 inch by 10.69 inch center-to-center spacing. The Region I Carborundum equipped rack has restrictive loading patterns to address degradation of neutron absorbing material in the rack. The loading patterns accommodate some face-adjacent fuel assemblies with consideration of burnup credit in Sub-Regions 1D and 1E. The Region 1 Metamic equipped racks are only restricted by maximum planar U^{235} enrichment. The Region I Carborundum equipped rack also has provisions for storing non-fissile bearing components.

Region II contains racks in both the spent fuel pool and the north tilt pit having a 9.17 inch center-to-center spacing. Because of the smaller spacing and an analyzed solid poison concentration of zero (Boraflex), Region II also has limitations for fuel storage. Further information on limitations can be found in Section 4.0, “Design Features.” These limitations (e.g., enrichment, burnup, loading patterns) are sufficient to maintain a keff of ≤ 0.95 when flooded with borated water and keff < 1.0 when flooded with unborated water.

APPLICABLE SAFETY ANALYSES

The fuel storage facility was originally designed for noncriticality by use of adequate spacing, and "flux trap" construction, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans. The current criticality calculations also take credit for soluble boron to prevent criticality.

The spent fuel pool storage meets the requirements specified in “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants”, Laurence I. Kopp, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Reactor Systems Branch, February 1998. This document

BASES

ACTIONS

When the configuration of fuel assemblies or non-fissile bearing components stored in the spent fuel pool is not in accordance with the storage requirements, immediate action must be taken to make the necessary movement(s) to bring the configuration into compliance with the requirements.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that the combination of fuel assembly maximum nominal planar average enrichment and proposed fuel assembly placement is in accordance with Specification 4.3.1.1 prior to placing the assembly in a Region I Carborundum equipped storage location. This SR also verifies by administrative means that non-fissile bearing component storage will be in accordance with Specification 4.3.1.1m prior to placing the component in a Region I Carborundum storage location.

This SR also verifies by administrative means that the nominal planar average enrichment is in accordance with Specification 4.3.1.2 prior to placing the assembly in a Region I Metamic equipped storage location.

This SR also verifies by administrative means that the combination of maximum nominal planar average U-235 enrichment, burnup and decay time of the fuel assembly is in accordance with Tables 3.7.16-1 through 3.7.16-5, as appropriate, in the accompanying LCO prior to placing the fuel assembly in a storage location in the Region I Carborundum equipped rack or the Region II racks.

REFERENCES

None

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Primary Coolant System (PCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives, and thus is indication of current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 0.3 gpm tube leak of primary coolant at the limit of 1.0 μCi/gm. The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and primary coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

Operating a plant at the allowable limits would result in a 2 hour Exclusion Area Boundary (EAB) exposure within applicable 10 CFR 50.67 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analysis of the Main Steam Line Break (MSLB), outside of containment as discussed in the FSAR, Chapter 14.14 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of a MSLB are well within the plant EAB limits (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES (continued)	With the loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through Main Steam Safety Valves (MSSVs) and Atmospheric Dump Valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generator. Venting continues until the primary coolant temperature and pressure have decreased sufficiently for the Shutdown Cooling System to complete the cooldown. In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through MSSVs and ADVs during the event. Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).
LCO	As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 limits the radiological consequences of a Design Basis Accident (DBA) to well within the required limit (Ref. 1). Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the plant in an operational MODE that would minimize the radiological consequences of a DBA.
APPLICABILITY	In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere. In MODES 5 and 6, the steam generators are not being used for heat removal. Both the PCS and steam generators are at low pressure or depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant is an indication of a problem in the PCS and contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in primary coolant activity or LEAKAGE. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50.67

2. FSAR, Section 14.14

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

Sources of AC power to the plant Class 1E Electrical Power Distribution System include the offsite power sources, and the Class 1E onsite standby power sources, Diesel Generators 1-1 and 1-2 (DGs). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The AC power system at Palisades consists of a 345 kV switchyard, three circuits connecting the plant with off-site power (station power, startup, and safeguards transformers), the on-site distribution system, and two DGs. The on-site distribution system is divided into safety related (Class 1E) and non-safety related portions.

The switchyard interconnects six transmission lines from the off-site transmission system, the output line from the Covert Generating Station, and the output line from the Palisades main generator. These lines are connected in a "breaker and a half" scheme between the Front (F) and Rear (R) buses such that any single off-site line may supply the Palisades station loads when the plant is shutdown.

Two circuits supplying Palisades 2400 V buses from off-site are fed directly from a switchyard bus through the startup and safeguards transformers. They are available both during operation and during shutdown. The third circuit supplies the plant loads by "back feeding" through the main generator output circuit and station power transformers after the generator has been disconnected by a motor operated disconnect.

The station power transformers are connected into the main generator output circuit. Station power transformers 1-1 and 1-2 connect to the generator 22 kV output bus. Station power transformer 1-3 connects to the generator output line on the high voltage side of the main transformer. Station power transformers 1-1 and 1-3 supply non-safety related 4160 V loads during plant power operation and during backfeeding operations. Station power transformer 1-2 can supply both safety related and non-safety related 2400 V loads during backfeeding operation.

BACKGROUND (continued)

The three startup transformers are connected to a common 345 kV overhead line from the switchyard R bus. Startup transformers 1-1 and 1-3 supply 4160 V non-safety related station loads; Startup Transformer 1-2 can supply both safety related and non-safety related 2400 V loads. The startup transformers are available during operation and shutdown.

Safeguards Transformer 1-1 is connected to the switchyard F bus. It feeds station 2400 V loads through an underground line. It is available to supply these loads during operation and shutdown.

The onsite distribution system consists of seven main distribution buses (4160 V buses 1A, 1B, 1F, and 1G, and 2400 V buses 1C, 1D, and 1E) and supported lower voltage buses, Motor Control Centers (MCCs), and lighting panels. The 4160 V buses and 2400 V bus 1E are not safety related. Buses 1C and 1D and their supported buses and MCCs form two independent, redundant, safety related distribution trains. Each distribution train supplies one train of engineered safety features equipment.

In the event of a generator trip, all loads supplied by the station power transformers are automatically transferred to the startup transformers. Loads supplied by the safeguards transformer are unaffected by a plant trip. If power is lost to the safeguards transformer, the 2400 V loads will automatically transfer to startup transformer 1-2. If the startup transformers are not energized when these transfers occur, their output breakers will be blocked from closing and the 2400 V safety related buses will be energized by the DGs.

The two DGs each supply one 2400 V bus. They provide backup power in the event of loss of off-site power, or loss of power to the associated 2400 V bus. The continuous rating of the DGs is 2500 kW, with 110 percent overload permissible for 2 hours. The required fuel in the Fuel Oil Storage Tank and DG Day Tank will supply one DG for a minimum period of 7 days assuming accident loading conditions.

If either 2400 V bus, 1C or 1D, experiences a sustained undervoltage, the associated DG is started, the affected bus is separated from its offsite power sources, major loads are stripped from that bus and its supported buses, the DGs are connected to the bus, and ECCS or shutdown loads are started by an automatic load sequencer.

BACKGROUND (continued)

The DGs share a common fuel oil storage and transfer system. A single buried Fuel Oil Storage Tank is used, along with an individual day tank for each DG, to maintain the required fuel oil inventory. Two fuel transfer pumps are provided. The fuel transfer pumps are necessary for long-term operation of the DGs. Testing and analysis have shown that each DG consumes about 200 gallons of fuel oil per hour at 2750 kW and about 180 gallons of fuel oil per hour at 2500 kW. Each day tank is required to contain at least 2500 gallons and contains sufficient fuel for about 13.5 hours of full load operation (Ref. 8). Beyond that time, a fuel transfer pump is required for continued DG operation.

Either fuel transfer pump is capable of supplying either DG. However, each fuel transfer pump is not capable, with normally available switching, of being powered from either DG. DG 1-1 can power either fuel transfer pump, but DG 1-2 can only power P-18A. The fuel oil pumps share a common fuel oil storage tank, and common piping.

Fuel transfer pump P-18A is powered from MCC-8, which is normally connected to Bus 1D (DG 1-2) through Station Power Transformer 12 and Load Center 12. In an emergency, P-18A can be powered from Bus 1C (DG 1-1) by cross-connecting Load Centers 11 and 12.

Fuel transfer pump P-18B is powered from MCC-1, which is normally connected to Bus 1C (DG 1-1) through Station Power Transformer 19 and Load Center 19. P-18B cannot be powered, using installed equipment, from Bus 1D (DG 1-2).

APPLICABLE SAFETY ANALYSES

The safety analyses do not explicitly address AC electrical power. They do, however, assume that the Engineered Safety Features (ESF) are available. The OPERABILITY of the ESF functions is supported by the AC Power Sources.

The design requirements are for each assumed safety function to be available under the following conditions:

a. The occurrence of an accident or transient,

b. The resultant consequential failures,

c. A worst-case single active failure,

d. Loss of all offsite or all onsite AC power, and

e. The most reactive control rod fails to insert.

APPLICABLE SAFETY ANALYSES (continued)

One proposed mechanism for the loss of off-site power is a perturbation of the transmission grid because of the loss of the plant's generating capacity. A loss of off-site power as a result of a generator trip can only occur during MODE 1 with the generator connected to the grid. However, it is also assumed in analysis for some events in MODE 2, such as a control rod ejection. No specific mechanism for initiating a loss of off-site power when the plant is not on the line is discussed in the FSAR.

In most cases, it is conservative to assume that off-site power is lost concurrent with the accident and that the single failure is that of a DG. That would leave only one train of safeguards equipment to cope with the accident, the other being disabled by the loss of AC power. Those analyses which assume that a loss of off-site power and failure of a single DG accompany the accident assume 11 seconds from the loss of power until the bus is re-energized. This time includes time for all portions of the circuitry necessary for detecting the undervoltage (relays and auxiliary relays) and starting the DG. Included in the 11 seconds, the analyses also assume 10 seconds for the DG to start and connect to the bus, and additional time for the sequencer to start each safeguards load.

The same assumptions are not conservative for all accident analyses. When analyzing the effects of a steam or feed line break, the loss of the condensate and feedwater pumps would reduce the steam generator inventory, so a loss of off-site power is not assumed.

In MODE 5 and MODE 6, loss of off-site power can be considered as an initiating event for a loss of shutdown cooling event.

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

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System and an independent DG for each safeguards train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence or a postulated DBA.

LCO
(continued)

General Design Criterion 17 (Ref. 1) requires, in part, that: "Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions."

The qualified offsite circuits available are Safeguards Transformer 1-1 and Startup Transformer 1-2. Station Power Transformer 1-2 is not qualified as a required source for LCO 3.8.1 since it is not independent of the other two offsite circuits. Station Power Transformer 1-2 will not be used in normal operations to power the 2400 V safety related buses in Modes 1-4.

Each offsite circuit must be capable of maintaining acceptable frequency and voltage, and accepting required loads during an accident, while supplying the 2400 V safety related buses.

Following a loss of offsite power, each DG must be capable of starting and connecting to its respective 2400 V bus. This will be accomplished within 10 seconds after receipt of a DG start signal. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the 2400 V safety related buses.

Proper sequencing of loads and tripping of nonessential loads are required functions for DG OPERABILITY.

APPLICABILITY

The AC sources are required to be OPERABLE above MODE 5 to ensure that redundant sources of off-site and on-site AC power are available to support engineered safeguards equipment in the event of an accident or transient. The AC sources also support the equipment necessary for power operation, plant heatups and cooldowns, and shutdown operation.

The AC source requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are addressed in LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable DG. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable DG and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

To ensure a highly reliable power source remains with the one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in failure to meet this Required Action. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

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 3.8.1 A.1 must be initially performed within 1 hour without any SR 3.0.2 extension, subsequent performances at the “Once per 8 hours” interval may utilize the 25% SR 3.0.2 extension.

A.2

According to the recommendations of Regulatory Guide (RG) 1.93 (Ref. 2), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the plant safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

ACTIONS

A.2 (continued)

The 72-hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. The second Completion Time for Required Action A.2 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single continuous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable, and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 7 days. This could lead to a total of 10 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional 7 days (for a total of 17 days) allowed prior to complete restoration of the LCO. The 10-day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and 10 day Completion Time means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

The Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

ACTIONS (continued)

B.2

In accordance with LCO 3.0.6, the requirement to declare required features inoperable carries with it the requirement to take those actions required by the LCO for that required equipment.

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. Redundant required feature failures consist of inoperable features within a train redundant to the train that has an inoperable DG. If the train that has an inoperable DG contains multiple features redundant to the inoperable feature in the other train, all those multiple features must be declared inoperable. For example, if DG 1-1 and Containment Spray Pump P-54A are inoperable concurrently, Containment Spray Pumps P-54B and P-54C must both be declared inoperable. In this example, if off-site power were lost, neither P-54B nor P-54C would be available.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature on the other train is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required supporting or supported features, or both, that are associated with the OPERABLE DG, results in starting the Completion Time for Required Action B.2. Four hours from the discovery of these events existing concurrently, is acceptable because it minimizes risk while allowing time for restoration before subjecting the plant to transients associated with shutdown.

In this Condition, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost.

ACTIONS

B.2 (continued)

The 4-hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4-hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3 provides an allowance to avoid unnecessary testing of the OPERABLE DG. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 (test starting of the OPERABLE DG) does not have to be performed. If the cause of inoperability exists on other DGs, the other DGs would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed to not exist on the remaining DG, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

In the event the inoperable DG is restored to OPERABLE status prior to completing Required Action B.3.1 or B.3.2 the corrective action system would normally continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24-hour constraint imposed while in Condition B. According to Generic Letter 84-15 (Ref. 3), 24 hours is reasonable to confirm that the OPERABLE DG is not affected by the same problem as the inoperable DG.

ACTIONS (continued)

B.4

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System for a limited period. The 7-day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently returned OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 10 days, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 13 days) allowed prior to complete restoration of the LCO. The 10-day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 7 day and 10 day Completion Time means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

ACTIONS (continued)

C.1

In accordance with LCO 3.0.6 the requirement to declare required features inoperable carries with it the requirement to take those actions required by the LCO for that required equipment.

Required Action C.1, which applies when two required offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours. The rationale for the reduction to 12 hours is that RG 1.93 (Ref. 2) recommends a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition C (two offsite circuits inoperable), a required feature becomes inoperable, this Completion Time begins to be tracked.

ACTIONS (continued)

C.2

According to the recommendations of RG 1.93 (Ref. 2), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to accomplish a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the plant in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst-case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, “Distribution Systems - Operating,” must be immediately entered. This allows Condition D to provide the requirements for the loss of one offsite circuit and one DG without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. The 12-hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

According to the recommendations of RG 1.93 (Ref. 2), operation may continue in Condition D for a period that should not exceed 12 hours.

ACTIONS (continued)

E.1

With both DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, no AC source would be available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since an inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to the recommendations of RG 1.93 (Ref. 2), with both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

F.1 and F.2

If the inoperable AC power sources cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to an operating condition 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.

G.1

Condition G corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 4). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of RG 1.9 (Ref. 5) and RG 1.137 (Ref. 6).

Where the SRs discussed herein specify voltage and frequency tolerances for the DGs operated in the "Unit" mode, the following is applicable. The minimum steady state output voltage of 2280 V is 95% of the nominal 2400 V generator rating. This value is above the setting of the primary undervoltage relays (127-1 and 127-2) and above the minimum analyzed acceptable bus voltage. It also allows for voltage drops to motors and other equipment down through the 120 V level. The specified maximum steady state output voltage of 2520 V is 105% of the nominal generator rating of 2400 V. It is below the maximum voltage rating of the safeguards motors, 2530 V. The specified minimum and maximum frequencies of the DG are 59.5 Hz and 61.2 Hz, respectively. The minimum value assures that ESF pumps provide sufficient flow to meet the accident analyses. The maximum value is equal to 102% of the 60 Hz nominal frequency and is derived from the recommendations given in RG 1.9 (Ref. 5).

Higher maximum tolerances are specified for final steady state voltage and frequency following a loss of load test, because that test must be performed with the DG controls in the "Parallel" mode. Since "Parallel" mode operation introduces both voltage and speed droop, the DG final conditions will not return to the nominal "Unit" mode settings.

SR 3.8.1.1

This SR assures that the required offsite circuits are OPERABLE. Each offsite circuit must be energized from associated switchyard bus through its disconnect switch to be OPERABLE.

Since each required offsite circuit transformer has only one possible source of power, the associated switchyard bus, and since loss of voltage to either the switchyard bus or the transformer is alarmed in the control room, correct alignment and voltage may be verified by the absence of these alarms.

SURVEILLANCE REQUIREMENTS

SR 3.8.1.1 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.2

This SR helps to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the plant in a safe shutdown condition.

The test starting of the DG provides assurance that the DG would start and be ready for loading in the time period assumed in the safety analyses. The test, however does not, and is not intended to, test all portions of the circuitry necessary for automatic starting and loading. The operation of the bus undervoltage relays and their auxiliary relays which initiate DG starting, the control relay, which initiates DG breaker closure, and the DG breaker closure itself are not verified by this test. Verification of automatic operation of these components requires de-energizing the associated 2400 V bus and cannot be done during plant operation. For this test, the 10-second timing is started when the DG receives a start signal, and ends when the DG voltage sensing relays actuate. For the purposes of SR 3.8.1.2, the DGs are manually started from standby conditions. Standby conditions for a DG mean the diesel engine is not running, its coolant and oil temperatures are being maintained consistent with manufacturer recommendations, and ≥ 20 minutes have elapsed since the last DG air roll.

Three relays sense the terminal voltage on each DG. These relays, in conjunction with a load shedding relay actuated by bus undervoltage, initiate automatic closing of the DG breaker. During testing, the actuation of the three voltage sensing relays is used as the timing point to determine when the DG is ready for loading.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads for at least 15 minutes. A minimum total run time of 60 minutes is required to stabilize engine temperatures.

During the period when the DG is paralleled to the grid, it must be considered inoperable. This is because there are no provisions to automatically shift the DG controls from parallel mode to unit mode. Additionally, when paralleled, there are certain conditions where the protection schemes may not prevent DG overloading and subsequent breaker trip and lockout.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by three Notes. Note 1 states that momentary transients outside the required band do not invalidate this test. This is to assure that a minor change in grid conditions and the resultant change in DG load, or a similar event, does not result in a surveillance being unnecessarily repeated. Note 2 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 3 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The specified level is adequate for a minimum of 13.5 hours of DG operation at full load.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.5

Each DG is provided with an engine overspeed trip to prevent damage to the engine. The loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. This Surveillance may be accomplished with the DG in the "Parallel" mode.

An acceptable method is to parallel the DG with the grid and load the DG to a load equal to or greater than its single largest post-accident load. The DG breaker is tripped while its voltage and frequency (or speed) are being recorded. The time, voltage, and frequency tolerances specified in this SR are derived from the recommendations of RG 1.9, Revision 3 (Ref. 5).

RG 1.9 (Ref. 5) recommends that the increase in diesel speed during the transient does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower. The Palisades DGs have a synchronous speed of 900 rpm and an overspeed trip setting range of 1060 to 1105 rpm. Therefore, the maximum acceptable transient frequency for this SR is 68 Hz.

The minimum steady state voltage is specified to provide adequate margin for the switchgear and for both the 2400 and 480 V safeguards motors; the maximum steady state voltage is 2400 +10% V as recommended by RG 1.9 (Ref. 5).

The minimum acceptable frequency is specified to assure that the safeguards pumps powered from the DG would supply adequate flow to meet the safety analyses. The maximum acceptable steady state frequency is slightly higher than the +2% (61.2 Hz) recommended by RG 1.9 (Ref. 5) because the test must be performed with the DG controls in the Parallel mode. The increased frequency allowance of 0.3 Hz is based on the expected speed differential associated with performance of the test while in the "Parallel" mode.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.6

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine and generator load response under a complete loss of load. These acceptance criteria provide DG damage protection. The 4000 V limitation is based on generator rating of 2400/4160V and the ratings of those components (connecting cables and switchgear) that would experience the voltage transient. While the DG is not expected to experience this transient during an event and continue to be available, this response ensures that the DG is not degraded for future application, including re-connection to the bus if the trip initiator can be corrected or isolated.

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, yet still provide adequate testing margin between the specified power factor limit and the DG design power factor limit of 0.8, testing must be performed using a power factor $\leq$ 0.9. This is consistent with RG 1.9 (Ref. 5).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.7

As recommended by RG 1.9 (Ref. 5) this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and re-energizing of the emergency buses and respective loads from the DG.

The requirement to energize permanently connected loads is met when the DG breaker closes, energizing its associated 2400 V bus. Permanently connected loads are those that are not disconnected from the bus by load shedding relays. They are energized when the DG breaker closes. It is not necessary to monitor each permanently connected load.

SURVEILLANCE REQUIREMENTS

SR 3.8.1.7 (continued)

The DG auto-start and breaker closure time of 10 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. For this test, the 10-second timing is started when the DG receives a start signal, and ends when the DG breaker closes. The safety analyses assume 11 seconds from the loss of power until the bus is re-energized.

The requirement to verify that auto-connected shutdown loads are energized refers to those loads that are actuated by the Normal Shutdown Sequencer. Each load should be started to assure that the DG is capable of accelerating these loads at the intervals programmed for the Normal Shutdown Sequence. The sequenced pumps may be operating on recirculation flow.

The requirements to maintain steady state voltage and frequency apply to the "steady state" period after all sequenced loads have been started. This period need only be long enough to achieve and measure steady voltage and frequency.

The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved. The requirement to supply permanently connected loads for ≥ 5 minutes, refers to the duration of the DG connection to the associated safeguards bus. It is not intended to require that sequenced loads be operated throughout the 5-minute period. It is not necessary to monitor each permanently connected load.

The requirement to verify the connection and supply of permanently and automatically connected loads is intended to demonstrate the DG loading logic. This testing may be accomplished in any series of sequential, overlapping, or total steps so that the required connection and loading sequence is verified.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.8

RG 1.9 (Ref. 5) recommends demonstration that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 120 minutes of which is at a load above its analyzed peak accident loading and the remainder of the time at a load equivalent to the continuous duty rating of the DG. SR 3.8.1.8 only requires ≥ 100 minutes at a load above the DG analyzed peak accident loading. The 100 minutes required by the SR satisfies the intent of the recommendations of the RG, but allows some tolerance between the time requirement and the DG rating. Without this tolerance, the load would have to be reduced at precisely 2 hours to satisfy the SR without exceeding the manufacturer's rating of the DG.

The DG starts for this Surveillance can be performed either from standby or hot conditions.

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, yet still provide adequate testing margin between the specified power factor limit and the DG design power factor limit of 0.8, testing must be performed using a power factor of ≤ 0.9. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

In addition, a Note to the SR states that momentary transients outside the required band do not invalidate this test. This is to assure that a minor change in grid conditions and the resultant change in DG load, or a similar event, does not result in a surveillance being unnecessarily repeated.

During the period when the DG is paralleled to the grid, it must be considered inoperable. This is because there are no provisions to automatically shift the DG controls from parallel mode to unit mode. Additionally, when paralleled, there are certain conditions where the protection schemes may not prevent DG overloading and subsequent breaker trip and lockout.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

As recommended by RG 1.9 (Ref. 5), this Surveillance ensures that the manual synchronization and load transfer from the DG to the offsite source can be made and that the DG can be returned to ready to load status when offsite power is restored. The test is performed while the DG is supplying its associated 2400 V bus, but not necessarily carrying the sequenced accident loads. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open, the automatic load sequencer is reset, and the DG controls are returned to "Unit."

During the period when the DG is paralleled to the grid, it must be considered inoperable. This is because there are no provisions to automatically shift the DG controls from parallel mode to unit mode. Additionally, when paralleled, there are certain conditions where the protection schemes may not prevent DG overloading and subsequent breaker trip and lockout.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.10

If power is lost to bus 1C or 1D, loads are sequentially connected to the bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs by concurrent motor starting currents. The 0.3-second load sequence time tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and ensures that safety analysis assumptions regarding ESF equipment time delays are met. Logic Drawing E-17 Sheet 4 (Ref. 7) provides a summary of the automatic loading of safety related buses.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.11

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, PCS, and containment design limits are not exceeded.

The requirement to energize permanently connected loads is met when the DG breaker closes, energizing its associated 2400 V bus. Permanently connected loads are those that are not disconnected from the bus by load shedding relays. They are energized when the DG breaker closes. It is not necessary to monitor each permanently connected load. The DG auto-start and breaker closure time of 10 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. For this test, the 10-second timing is started when the DG receives a start signal, and ends when the DG breaker closes. The safety analyses assume 11 seconds from the loss of power until the bus is re-energized.

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

In addition, a Note to the SR states that momentary transients outside the required band do not invalidate this test. This is to assure that a minor change in grid conditions and the resultant change in DG load, or a similar event, does not result in a surveillance being unnecessarily repeated.

The requirement to verify that auto-connected shutdown loads are energized refers to those loads that are actuated by the DBA Sequencer. Each load should be started to assure that the DG is capable of accelerating these loads at the intervals programmed for the DBA Sequence. Since the containment spray pumps do not actuate on SIS generated by Pressure Low Pressure, the test should be performed such that spray pump starting by the sequencer is also verified along with the other SIS loads. The sequenced pumps may be operating on recirculation flow or in other testing modes. The requirements to maintain steady state voltage and frequency apply to the "steady state" period after all sequenced loads have been started. This period need only be long enough to achieve and measure steady voltage and frequency.

The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved. The requirement to supply permanently connected loads for ≥ 5 minutes, refers to the duration of the DG connection to the associated 2400 V bus. It is not intended to require that sequenced loads be operated throughout the 5-minute period. It is not necessary to monitor each permanently connected load.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17
2. Regulatory Guide 1.93, December 1974
3. Generic Letter 84-15, July 2, 1984
4. 10 CFR 50, Appendix A, GDC 18
5. Regulatory Guide 1.9, Rev. 3, July 1993
6. Regulatory Guide 1.137, Rev. 1, October 1979
7. Palisades Logic Drawing E-17, Sheet 4
8. Engineering Change 12118

INSERT Bases 3.8.2

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The safety analyses do not explicitly address electrical power. They do, however, assume that various electrically powered and controlled equipment is available. Electrical power is necessary to terminate and mitigate the effects of many postulated events which could occur in MODES 5 and 6.

Analyzed events which might occur during MODES 5 and 6 are Loss of PCS inventory or Loss of PCS Flow, (which in MODES 5 and 6 would be grouped as a Loss of Shutdown Cooling event), and radioactive releases (Fuel Handling Accident, Cask Drop, Radioactive Gas Release, Etc.).

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed above MODE 5 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the primary coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced, and in minimal consequences.

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

LCO

This LCO requires one offsite circuit to be OPERABLE. One OPERABLE offsite circuit ensures that all required loads may be powered from offsite power. Since only one offsite AC source is required, independence is not a criterion. Any of the three offsite supplies, Safeguards Transformer 1-1, Station Power Transformer 1-2, or Startup Transformer 1-2 is acceptable as a qualified circuit.

LCO (continued)

An OPERABLE DG, associated with a distribution subsystem required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit.

Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and loss of shutdown cooling).

The DG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective 2400 V bus on detection of bus undervoltage, and accepting required loads. Proper "Normal Shutdown" loading sequence, and tripping of nonessential loads, is a required function for DG OPERABILITY. A Service Water Pump must be started soon after the DG to assure continued DG operability. The DBA loading sequence is not required to be OPERABLE since the Safety Injection Signal is disabled during MODES 5 and 6.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that equipment and instrumentation is available to:

a. Provide coolant inventory makeup,

b. Mitigate a fuel handling accident,

c. Mitigate shutdown events that can lead to core damage, and

d. Monitor and maintain the plant in a cold shutdown condition or refueling condition.

This LCO is applicable during movement of irradiated fuel assemblies even if the plant is in a condition other than MODES 5 and 6. This LCO provides the necessary ACTIONS if the AC electrical power sources required by this LCO become unavailable during movement of irradiated fuel assemblies.

The AC source requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.1, "AC Sources - Operating."

ACTIONS

A.1

An offsite circuit would be considered inoperable if it were not available to supply the 2400 V safety related bus or buses required by LCO 3.8.10. Since the required offsite AC source is only required to support features required by other LCOs, the option to declare those required features with no offsite power available to be inoperable, assures that appropriate ACTIONS will be implemented in accordance with the affected LCOs.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

Required Action A.1 may involve undesired and unnecessary administrative efforts, therefore, Required Action A.2.1 through A.2.4 provide alternate, but sufficiently conservative, actions for unplanned losses of AC sources.

With the required DG inoperable, the minimum required diversity of AC power sources is not available.

Required Actions A.2.1 through A.2.4, and B.1 through B.4 require suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The suspension of CORE ALTERATIONS and movement of irradiated fuel assemblies does not preclude actions to place a fuel assembly in a safe location; the suspension of positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SHUTDOWN MARGIN is maintained.

These ACTIONS minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources (and to continue this action until restoration is accomplished) in order to provide the necessary AC power to the plant safety systems.

The Completion Time of "immediately" is consistent with the required times for actions requiring prompt attention. The restoration of the required AC power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.8.2.1 SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in MODES 5 and 6. The SRs from LCO 3.8.1 which are required are those which both support a feature required in MODES 5 and 6 and which can be performed without affecting the OPERABILITY or reliability of the required sources. With only one DG available, many tests cannot be performed since their performance would render that DG inoperable during the test. This is the case for tests which require DG loading: SRs 3.8.1.3, 3.8.1.5, 3.8.1.6, 3.8.1.7, 3.8.1.8, 3.8.1.9, 3.8.1.10, and 3.8.1.11.
REFERENCES	None

BASES

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel, Lube Oil, and Starting Air

BASES

BACKGROUND

The Diesel Generators (DGs) are provided with a storage subsystem having a required fuel oil inventory sufficient to operate one diesel for a period of 7 days, while the DG is supplying maximum post-accident loads. The fuel oil storage subsystem is comprised of the Fuel Oil Storage Tank and a fuel oil day tank. This onsite fuel oil capacity is sufficient to operate the DG for longer than the time to replenish the onsite supply from offsite sources.

Fuel oil is transferred from the Fuel Oil Storage Tank to either day tank by either of two Fuel Transfer Systems. The fuel oil transfer system which includes fuel transfer pump P-18A can be powered by offsite power, or by either DG. However, the fuel oil transfer system which includes fuel transfer pump P-18B can only be powered by offsite power, or by DG 1-1.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide (RG) 1.137 (Ref. 1) addresses the recommended fuel oil practices as supplemented by ANSI N195-1976 (Ref. 2).

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. The onsite storage is sufficient to ensure 7 days of continuous operation. This supply is sufficient supply to allow the operator to replenish lube oil from offsite sources. Implicit in this LCO is the requirement to assure, though not necessarily by testing, the capability to transfer the lube oil from its storage location to the DG oil sump, while the DG is running.

Each DG is provided with an associated starting air subsystem to assure independent start capability. The starting air system is required to have a minimum capacity with margin for a DG start attempt without recharging the air start receivers.

BASES

APPLICABLE SAFETY ANALYSES

A description of the Safety Analyses applicable in MODES 1, 2, 3, and 4 is provided in the Bases for LCO 3.8.1, "AC Sources - Operating"; during MODES 5 and 6, in the Bases for LCO 3.8.2, "AC Sources - Shutdown." Since diesel fuel, lube oil, and starting air subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of full accident load operation. It is also required to meet specific standards for quality. Additionally, the ability to transfer fuel oil from the storage tank to each day tank is required from each of the two transfer pumps.

Additionally, sufficient lube oil supply must be available to ensure the capability to operate at full accident load for 7 days. This requirement is in addition to the lube oil contained in the engine sump.

The starting air subsystem must provide, without the aid of the refill compressor, sufficient air start capacity, including margin, to assure start capability for its associated DG.

These requirements, in conjunction with an ability to obtain replacement supplies within 7 days, support the availability of the DGs. DG day tank fuel requirements are addressed in LCOs 3.8.1 and 3.8.2.

APPLICABILITY

DG OPERABILITY is required by LCOs 3.8.1 and 3.8.2 to ensure the availability of the required AC power to shut down the reactor and maintain it in a safe shutdown condition following a loss of off-site power. Since diesel fuel, lube oil, and starting air support LCOs 3.8.1 and 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits, and the fuel transfer system is required to be OPERABLE, when either DG is required to be OPERABLE.

ACTIONS

A.1

In this Condition, the available DG fuel oil supply is less than the required 7 day supply, but enough for at least 6 days. The fuel oil inventory equivalent to a 6 day supply is 28,592 gallons (Ref. 5). This inventory is conservatively based on an uprated 2600 kW DG capacity. This condition allows sufficient time to obtain additional fuel and to perform the sampling and analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required inventory prior to declaring the DGs inoperable.

B.1

In this Condition, the available DG lube oil supply in storage is less than the required 7 day supply, but enough for at least 6 days. The lube oil inventory equivalent to a 6 day supply is 268 gallons (Ref. 5). This inventory is conservatively based on an uprated 2600 kW DG capacity. This condition allows sufficient time to obtain additional lube oil. A period of 48 hours is considered sufficient to complete restoration of the required inventory prior to declaring the DGs inoperable.

C.1, D.1, and E.1

Fuel transfer pump P-18A can be controlled either manually or automatically via fuel oil day tank level controls. Fuel transfer pump P-18B can only be controlled manually. The P-18A fuel transfer system is OPERABLE if fuel oil can be transferred either automatically or manually by the P-18A fuel transfer pump.

If a fuel oil system is incapable of supplying fuel oil to a day tank or an engine mounted tank as required, then only the DG associated with that day tank or engine mounted tank is required to be declared inoperable if the fuel transfer system is not restored to OPERABLE status within its specified Completion Time.

Since DG 1-2 cannot power fuel transfer pump P-18B, without P-18A, DG 1-2 becomes dependent on offsite power or DG 1-1 for its fuel supply (beyond the approximately 13.5 hours it will operate on the day tank), and does not meet the requirement for independence. Since the condition is not as severe as the DG itself being inoperable, 12 hours is allowed to restore the fuel transfer system to operable status prior to declaring the DG inoperable.

Without P-18B, either DG can still provide power to the remaining fuel transfer system. Therefore, neither DG is directly affected. Continued operation with a single remaining fuel transfer system, however, must

ACTIONS

C.1, D.1, and E.1 (continued)

be limited since an additional single active failure (P-18A) could disable the onsite power system. Because the loss of P-18B is less severe than the loss of one DG, a 7 day Completion Time is allowed.

If both fuel transfer systems are inoperable, the onsite AC sources are limited to about 13.5 hours duration. Since this condition is not as severe as both DGs being inoperable, 8 hours is allowed to restore one fuel transfer pump to OPERABLE status.

F.1

With the stored fuel oil properties, other than viscosity, and water and sediment, defined in the Fuel Oil Testing Program not within the required limits, but acceptable for short term DG operation, a period of 30 days is allowed for restoring the stored fuel oil properties. The most likely cause of stored fuel oil becoming out of limits is the addition of new fuel oil with properties that do not meet all of the limits. This 30 day period provides sufficient time to determine if new fuel oil, when mixed with stored fuel oil, will produce an acceptable mixture, or if other methods to restore the stored fuel oil properties are required. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

G.1

With a Required Action and associated Completion Time not met, or with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than addressed by Conditions A, B, or F, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

In the event that diesel fuel oil with viscosity, or water and sediment is out of limits, this would be unacceptable for even short term DG operation. Viscosity is important primarily because of its effect on the handling of the fuel by the pump and injector system; water and sediment provides an indication of fuel contamination. When the fuel oil stored in the Fuel Oil Storage Tank is determined to be out of viscosity, or water and sediment limits, the DGs must be declared inoperable, immediately.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage subsystem to support either DG's operation for 7 days at full post-accident load. The fuel oil inventory equivalent to a 7 day supply is 33,054 gallons (Ref. 5) when calculated in accordance with References 1 and 2. This inventory is conservatively based on an uprated 2600 kW DG capacity. The required fuel storage volume is determined using the most limiting energy content of the stored fuel. Using the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel generator output, and the corresponding fuel consumption rate, the onsite fuel storage volume required for 7 days of operation can be determined. SR 3.8.3.3 requires new fuel to be tested to verify that the absolute specific gravity or API gravity is not less than the value assumed in the diesel fuel oil consumption calculations. The 7 day period is sufficient time to place the plant in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.3.2

This Surveillance ensures that sufficient stored lube oil inventory is available to support at least 7 days of full accident load operation for one DG. The lube oil inventory equivalent to a 7 day supply is 313 gallons and is based on an estimated consumption of 1.0% of fuel oil consumption (Ref. 5). This inventory is also conservatively based on an uprated 2600 kW DG capacity.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil and stored fuel oil are of the appropriate grade and have not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion.

Testing for viscosity, specific gravity, and water and sediment is completed for fuel oil delivered to the plant prior to its being added to the Fuel Oil Storage Tank. Fuel oil which fails the test, but has not been

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

added to the Fuel Oil Storage Tank does not imply failure of this SR and requires no specific action. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tank without concern for contaminating the entire volume of fuel oil in the storage tank.

Fuel oil is tested for other of the parameters specified in ASTM D975 (Ref. 3) in accordance with the Fuel Oil Testing Program required by Specification 5.5.11. Fuel oil determined to have one or more measured parameters, other than viscosity or water and sediment, outside acceptable limits will be evaluated for its effect on DG operation. Fuel oil which is determined to be acceptable for short term DG operation, but outside limits will be restored to within limits in accordance with LCO 3.8.3 Condition F.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The pressure specified in this SR is intended to reflect the acceptable margin from which successful starts can be accomplished.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the Fuel Oil Storage Tank eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it reduces the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.5 (continued)

The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed in accordance with the requirements of the Fuel Oil Testing Program.

SR 3.8.3.6

This SR demonstrates that the fuel transfer systems can, as applicable, automatically and manually transfer fuel from the Fuel Oil Storage Tank to each day tank, and automatically from each day tank to each engine mounted tank. Automatic or manual transfer of fuel oil is required to support continuous operation of standby power sources.

This SR provides assurance that the following portions of the fuel transfer system are OPERABLE:

a. Fuel transfer pumps;

b. Day and engine mounted tank filling solenoid valves;

c. Day tank fill via automatic level controls or manual operation; and

d. Engine mounted tank fill via automatic level controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Regulatory Guide 1.137

2. ANSI N195-1976

3. ASTM Standards, D975, Table 1

4. ASME, Boiler and Pressure Vessel Code, Section XI

5. Engineering Analysis EA-EC6432-01

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC power system with control power. It also provides control power to selected safety related equipment and power to the preferred AC Buses (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure.

The 125 V DC electrical power system consists of two independent and redundant safety related Class 1E DC power sources. Each DC source consists of one 125 V battery, one battery charger, and the associated control equipment and interconnecting cabling. While each station battery has two associated battery chargers, one powered by the associated AC power distribution system (the directly connected chargers), and one powered from the opposite AC power distribution system (the cross connected chargers), the cross connect chargers are not required to be OPERABLE and cannot be credited to meet this LCO. The battery chargers are normally operated in pairs, either both direct connected chargers or both cross connected chargers, to assure a diverse AC supply.

During normal operation, the 125 V DC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power from the battery charger, the DC load continues to be powered from the station batteries.

The DC power distribution system is described in the Bases for LCO 3.8.9, "Distributions System - Operating."

Each battery has adequate storage capacity to carry the required load continuously for at least 4 hours as discussed in the FSAR, Chapter 8 (Ref. 2).

Each 125 V battery is separately housed in a ventilated room apart from its charger and distribution centers. Each DC source is separated physically and electrically from the other DC source to ensure that a single failure in one source does not cause a failure in a redundant source.

BASES

BACKGROUND (continued)	The batteries for the DC power sources are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 125.7 V per battery discussed in the FSAR, Chapter 8 (Ref. 2). The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 3). Each DC electrical power source has ample power output capacity for the steady state operation of connected loads during normal operation, while at the same time maintaining its battery fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads discussed in the FSAR, Chapter 8 (Ref. 2).
APPLICABLE SAFETY ANALYSES	A description of the Safety Analyses applicable in MODES 1, 2, 3, and 4 is provided in the Bases for LCO 3.8.1, "AC Sources - Operating." The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	The DC power sources, each consisting of one battery, one directly connected battery charger and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of DC control power and Preferred AC power to shut down the reactor and maintain it in a safe condition. An OPERABLE DC electrical power source requires its battery to be OPERABLE and connected to the associated DC bus. In order for the battery to remain OPERABLE for any extended period of time, at least one charger must be in service. Without a charger in service, the DC loads would reduce the battery charge to the point where the battery would become inoperable. Disconnecting a charger, however, does not, in itself, make a battery inoperable. The LCO requires chargers ED-15 and ED-16 because those chargers are powered by the AC power distribution system and DG associated with the battery they supply. If only the cross connected chargers were available, and a loss of off-site power should occur concurrently with the loss of one DG, both safeguards trains would eventually become disabled. One train would be disabled by the lack of AC motive power;

BASES

LCO (continued)

the other would become disabled when the battery, whose only OPERABLE charger is fed by the failed DG, became depleted.

The required chargers, ED-15 and ED-16, must be OPERABLE, but need not actually be in service because the probability of a concurrent loss of offsite power with loss of one DG is low, and battery charging current is not needed immediately after an accident.

APPLICABILITY

The DC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that redundant sources of DC power are available to support engineered safeguards equipment and plant instrumentation in the event of an accident or transient. The DC sources also support the equipment and instrumentation necessary for power operation, plant heatups and cooldowns, and shutdown operation.

The DC source requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are addressed in LCO 3.8.5, "DC Sources - Shutdown."

ACTIONS

A.1 and A.2

With one of the required chargers (ED-15 or ED-16) inoperable, the cross-connected charger must be placed in service within 2 hours, if it is not already in service, to maintain the battery in OPERABLE status. If the cross-connect charger is not placed in service within 2 hours, Condition C would be entered.

Additionally for the cross-connected charger to be considered “functional,” the cross-connected charger must have been surveilled and satisfied the same performance test required for the directly connected charger (i.e., SR 3.8.4.6) within the required Frequency.

In order to limit the time when the DC source is not capable of continuously meeting the single failure criterion, the required charger must be restored to OPERABLE status within 7 days.

The 7 day Completion Time was chosen to allow trouble shooting, location of parts, and repair.

BASES

ACTIONS (continued)

B.1 and B.2

With one battery inoperable, the associated DC system cannot meet its design. It lacks both the surge capacity and the independence from AC power sources which the battery provides if offsite power is lost. Placing the second battery charger in service provides two benefits: 1) restoration of the capacity to supply a sudden DC power demand, and 2) restoration of adequate DC power in the affected train as soon as either AC power distribution system is re-energized following a loss of offsite power. If the cross-connect charger is not placed in service within 2 hours, Condition C would be entered. Additionally for the cross-connected charger to be considered “functional,” the cross-connected charger must have been surveilled and satisfied the same performance test required for the directly connected charger (i.e., SR 3.8.4.6) within the required Frequency.

In order to restore the DC source to its design capability, the battery must be restored to OPERABLE status within 24 hours. The 24 hour Completion Time is a feature of the original Palisades licensing basis and reflects the availability to provide two trains of DC power from either AC distribution system. Furthermore, it provides a reasonable time to assess plant status as a function of the inoperable DC electrical power source and, if the battery is not restored to OPERABLE status, to prepare to effect an orderly and safe plant shutdown.

C.1 and C.2

If the inoperable DC electrical power source cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to an operating condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous current required to overcome the internal losses of a battery and maintain the battery in a fully charged state. The specified voltage is the nominal rating of the battery. Surveillance voltage measurements may be adjusted for cable losses and for installed plant instrumentation to ensure that battery terminal voltage requirements are satisfied. At that terminal voltage, the battery has sufficient charge to provide the analyzed capacity for either accident loading or station blackout loading. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.4.2

Visual inspection to detect corrosion of the battery terminals and connectors, or measurement of the resistance of each inter-cell and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The specified limits of ≤ 50 μohm for inter-cell connections and terminal connections, and ≤ 360 μohms for inter-tier and inter-rack connections are in accordance with the manufacturers recommendations. The 50 μohm value is based on the minimum battery design voltage. Battery sizing calculations show the first minute load on the ED-02 battery as the load that determines battery size, hence, battery voltage will be at its lowest value while the battery supplies this current. Calculations also show that at a minimum temperature and end of life (80% battery performance), battery voltage during this first minute load will be about 1.815 V per cell, assuming nominal connection resistance. But if all the connections were at the ceiling value of 50 μohms, the battery manufacturer indicates that the additional voltage drop would result in a battery voltage of about 1.79 V per cell, which is still above the minimum design voltage (Ref. 5).

The 360 μohm value is based on 120% of the nominal cumulative resistance of the components which make up the connections: resistance of the connecting cable, and for each end of the cable, the battery post to cable lug connection, the cable lug itself, and the lug to cable connection.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.2 (continued)

The resistance values determined during initial battery installation are recorded with the battery replacement specifications, FES 95-206-ED-01 and FES 95-206-ED-02.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of inter-cell and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.4.

The specified limits for connection resistance are discussed in the Bases for SR 3.8.4.2.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.6

This SR requires that each required battery charger be capable of supplying 180 amps at 125 V for ≥ 8 hours. These requirements are based on the design capacity of the chargers. The chargers are rated at 200 amps; the specified 180 amps provides margin between the charger rating and the test requirement.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.4.7

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in FSAR Chapter 8 (Ref. 2).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.7 (continued)

The reason for the restriction that the plant be outside of MODES 1, 2, 3, and 4 is that performing the Surveillance requires disconnecting the battery from the DC distribution buses and connecting it to a test load resistor bank. This action makes the battery inoperable and completely unavailable for use.

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the "as found" condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

The modified performance discharge test is a simulated duty cycle that envelopes the Service Test Profile, is approved by the battery manufacturer, and is consistent with IEEE Standards. Since the ampere-hours removed by the initial loads represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified performance discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

The acceptance criteria for this Surveillance are consistent with the recommendations of IEEE-450 (Ref. 4) and IEEE-485 (Ref. 3). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation is indicated, according to IEEE-450 (Ref. 4), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 4).

The reason for the restriction that the plant be outside of MODES 1, 2, 3, and 4 is that performing the Surveillance requires disconnecting the battery from the DC distribution buses and connecting it to a test load resistor bank. This action makes the battery inoperable and completely unavailable for use.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17
2. FSAR, Chapter 8
3. IEEE-485-1983, June 1983
4. IEEE-450-1995
5. Letter; Graham Walker, C&D Charter Power Systems, Inc to John Slinkard, Consumers Power Company, 12 July 1996
6. Regulatory Guide 1.32, February 1977
7. Regulatory Guide 1.129, December 1974

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND	A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."
APPLICABLE SAFETY ANALYSES	A description of the Safety Analyses applicable during MODES 5 and 6 is provided in the Bases for LCO 3.8.2, "AC Sources - Shutdown." The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	This LCO requires those, and only those, DC power sources which supply the DC distribution subsystems required by LCO 3.8.10, to be OPERABLE. Each DC source consists of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling. The directly connected battery charger associated with the Diesel Generator (DG) required by LCO 3.8.2, “AC Sources - Shutdown” shall be OPERABLE. This ensures the availability of sufficient DC power sources to maintain the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and loss of shutdown cooling). Two DC sources are required to ensure availability of an uninterruptable power source for the instrumentation and controls for the Reactor Protective System (RPS). If the RPS is not required, then the LCO may continue to be met with one DC source provided: a. The OPERABLE DC source is associated with the DG required by LCO 3.8.2, “AC Sources - Shutdown”, and b. The cross connected battery charger (powered from the DG required by LCO 3.8.2, “AC Sources - Shutdown”) to the alternate bus is OPERABLE and in service. When an OPERABLE DC electrical power source requires its battery to be OPERABLE, it shall be connected to the associated DC bus. In order for the battery to remain OPERABLE for any extended period of time, at least one charger must be in service. Without a charger in service, the DC loads would reduce the battery charge to the point where the battery would become inoperable. Disconnecting a charger, however, does not, in itself, make a battery inoperable.

BASES

LCO (continued)

The required directly connected charger, ED-15 or ED-16, must be OPERABLE, but need not actually be in service because battery charging current is not needed immediately after an accident.

APPLICABILITY

The DC power sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that equipment and instrumentation is available to:

a. Provide coolant inventory makeup,

b. Mitigate a fuel handling accident,

c. Mitigate shutdown events that can lead to core damage, and

d. Monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

This LCO is applicable during movement of irradiated fuel assemblies even if the plant is in a condition other than MODE 5 or 6. This LCO provides the necessary ACTIONS if the DC electrical power sources required by this LCO become unavailable during movement of irradiated fuel assemblies.

The DC source requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.4, "DC Sources - Operating."

ACTIONS

A.1

Since the required DC source is only required to support features required by other LCOs, the option to declare those required features with no DC power available to be inoperable, assures that appropriate ACTIONS will be implemented in accordance with the affected LCOs.

A.2.1, A.2.2, A.2.3, and A.2.4

Required Action A.1 may involve undesired and unnecessary administrative efforts, therefore, Required Actions A.2.1 through A.2.4 provide alternate, but sufficiently conservative, actions for unplanned losses of a required DC source.

Required Actions A.2.1 through A.2.4 require suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The suspension of

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

CORE ALTERATIONS and movement of irradiated fuel assemblies does not preclude actions to place a fuel assembly in a safe location; the suspension of positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SHUTDOWN MARGIN is maintained.

These ACTIONS minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC sources (and to continue this action until restoration is accomplished) in order to provide the necessary DC power to the plant safety systems.

The Completion Time of "immediately" is consistent with the required times for actions requiring prompt attention. The restoration of the required DC power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient control and Preferred AC power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires the SRs from LCO 3.8.4 that are necessary for ensuring the OPERABILITY of the AC sources in MODES 5 and 6.

The SRs from LCO 3.8.4 which are required are those which can be performed without affecting the OPERABILITY or reliability of the required DC source. With only one battery available, loading tests cannot be performed since their performance would render that battery inoperable during the test. This is the case for SRs 3.8.4.7 and 3.8.4.8.

REFERENCES

None

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND	This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries. A discussion of these batteries is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."
APPLICABLE SAFETY ANALYSES	A description of the Safety Analyses applicable for MODES 1, 2, 3, and 4 is provided in the Bases for LCO 3.8.1, "AC Sources - Operating"; during MODES 5 and 6, in the Bases for LCO 3.8.2, "AC Sources - Shutdown." Battery cell parameters satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Battery cell limits are conservatively established, allowing continued DC electrical system function even when Category A and B limits are not met. The requirement to maintain the average temperature of representative cells above 70°F assures that the battery temperature is within the design band. Battery capacity is a function of battery temperature.
APPLICABILITY	The battery cell parameters are required solely for the support of the associated DC power sources. Therefore, they are only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussions in the Bases for LCO 3.8.4 and LCO 3.8.5, "DC Sources - Shutdown."

BASES

ACTIONS

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

With one or more cells in one or more batteries not within Category A or B limits but within the Category C limits, the battery is not fully charged but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be declared to be inoperable and continued operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cells.

Verification that all cells meet the Category C limits (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery will still be capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because specific gravity measurements may be required to be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A and B limits.

Battery cell parameters must be restored to Category A and B limits within 31 days.

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 3.8.6 A.2 must be initially performed within 24 hours without any SR 3.0.2 extension, subsequent performances at the “Once per 7 days” interval may utilize the 25% SR 3.0.2 extension.

BASES

ACTIONS (continued)

B.1

With the temperature of representative cells below the design temperature, or with one or more battery cells with parameters outside the Category C limits, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable.

Additionally, if battery cells cannot be restored to meeting Category A or B limits within 31 days, a serious difficulty with the battery is indicated and the battery must be declared to be inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 1), which recommends regular battery inspections including voltage, specific gravity, and electrolyte temperature of pilot cells. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.6.2

This Surveillance verification that the average temperature of representative cells is ≥ 70°F is consistent with a recommendation of IEEE-450 (Ref. 1). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

SR 3.8.6.3

The inspection of specific gravity and voltage is consistent with the recommendations of IEEE-450 (Ref. 1). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. Each category is discussed below.

Category A defines the fully charged parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage and specific gravity approximate the state of charge of the entire battery.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category A and B limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 1), with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote (a) to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A and B limit specified for float voltage is ≥ 2.13 V per cell. This value is based on a recommendation of IEEE-450, which states that prolonged operation of cells < 2.13 V can reduce their life expectancy.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.205. This value is six points (0.006) below the average baseline specific gravity for fully charged cells when the battery was installed and is characteristic of a charged cell with adequate capacity. The Category B limit specified for specific gravity for each connected cell is ≥ 1.200. Category B also requires that the average of all cells be ≥ 1.205 (0.006 below the baseline average of all cells). This allows some cells to be slightly lower than the nominal requirement as long as others are sufficiently higher so as to maintain the average above the nominal full charged value. According to IEEE-450, specific gravity readings are based on a temperature of 77°F (25°C).

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

Category C defines the limit for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limit, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limit specified for electrolyte level (above the top of the plates and not overflowing) ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limit for float voltage is based on IEEE-450, which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C specific gravity limit that each connected cell must be no less than 0.020 below the average of all connected cells and that average be ≥ 1.195 is based on manufacturer recommendations (0.020 below the manufacturer recommended fully charged, nominal specific gravity) (Ref. 2). This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

Footnote (a) allows for the normally observed level increase which occurs during sustained battery charging. Footnotes (b) and (c) to Table 3.8.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is < 2 amps on float charge. This current provides, in general, an indication of overall battery condition.

Footnote (c) to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity readings. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 1).

INSERT Bases 3.8.6

BASES

REFERENCES

1. IEEE-450-1995

2. C & D Standby Battery Installation and Operation Instructions

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND	The inverters (ED-06, ED-07, ED-08, and ED-09) are the normal source of power for the Preferred AC buses. The function of the inverter is to provide continuous AC electrical power to the Preferred AC buses, even in the event of an interruption to the normal AC power distribution system. A Preferred AC bus can be powered from the AC power distribution system via the Bypass Regulator if its associated inverter is out of service. An interlock prevents supplying more than one Preferred AC bus from the bypass regulator at any time. The station battery provides an uninterruptable power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Features (ESF).
APPLICABLE SAFETY ANALYSES	A description of the Safety Analyses applicable in MODES 1, 2, 3, and 4 is provided in the Bases for LCO 3.8.1, "AC Sources - Operating." Inverters are part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	The inverters ensure the availability of Preferred AC power for the instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Maintaining the inverters OPERABLE ensures that the redundancy incorporated into the RPS and ESF instrumentation and controls is maintained. The four inverters ensure an uninterruptable supply of AC electrical power to the Preferred AC buses even if the 2400 V safety related buses are de-energized. An inverter is considered inoperable if it is not powering the associated Preferred AC bus, or if its output voltage or frequency is not within tolerances.

BASES

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that redundant sources of Preferred AC power for instrumentation and control are available to support engineered safeguards equipment in the event of an accident or transient and for power operation, plant heatups and cooldowns, and shutdown operation.

Inverter requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are addressed in LCO 3.8.8, "Inverters - Shutdown."

ACTIONS

A.1

With an inverter inoperable, its associated Preferred AC bus becomes inoperable until it is manually re-energized from the bypass regulator. An inoperable Preferred AC Bus is addressed in LCO 3.8.9.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the plant is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail.

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to an operating condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.8.7.1 This Surveillance verifies that the inverters are functioning properly and energizing the Preferred AC buses. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESF connected to the Preferred AC buses. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
REFERENCES	None

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND	A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."
APPLICABLE SAFETY ANALYSES	A description of the Safety Analyses applicable during MODES 5 and 6 is provided in the Bases for LCO 3.8.2, "AC Sources - Shutdown." Inverters are part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	This LCO requires those, and only those, inverters necessary to support the Preferred AC buses required by LCO 3.8.10, "Distribution Systems - Shutdown," to be OPERABLE. As a minimum, both inverters associated with the Diesel Generator (DG) required by LCO 3.8.2, "AC Sources - Shutdown" shall be OPERABLE. This ensures required instrumentation and control functions are maintained by providing an uninterruptable supply of AC electrical power to those Preferred AC buses even if the 2400 V safety related bus is de-energized. If the Reactor Protective System (RPS) is required, then all four inverters shall be OPERABLE. This ensures the availability of sufficient Preferred AC electrical power to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and loss of shutdown cooling). An inverter is considered inoperable if it is not powering the associated Preferred AC bus, or if its voltage or frequency is not within tolerances.
APPLICABILITY	The inverters required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that equipment and instrumentation is available to: a. Provide coolant inventory makeup, b. Mitigate a fuel handling accident, c. Mitigate shutdown events that can lead to core damage, and

INSERT Bases 3.8.8

BASES

APPLICABILITY (continued)

d. Monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

This LCO is applicable during movement of irradiated fuel assemblies even if the plant is in a condition other than MODE 5 or 6. This LCO provides the necessary ACTIONS if the inverters required by this LCO become unavailable during movement of irradiated fuel assemblies.

Inverter requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.7.

ACTIONS

A.1

With a required inverter inoperable, its associated Preferred AC bus becomes inoperable until it is manually re-energized from the bypass regulator. An inoperable Preferred AC Bus is addressed in LCO 3.8.10.

A required inverter would be considered inoperable if it were not available to supply its associated Preferred AC bus. Since the inverter and its associated Preferred AC Bus is only required to support features required by other LCOs, the option to declare those required features without inverter supplied Preferred AC power available to be inoperable, assures that appropriate ACTIONS will be implemented in accordance with the affected LCOs.

A.2.1, A.2.2, A.2.3, and A.2.4

Required Action A.1 may involve undesired and unnecessary administrative efforts, therefore, Required Actions A.2.1 through A.2.4 provide alternate, but sufficiently conservative, actions for unplanned losses of required inverters.

Required Actions A.2.1 through A.2.4 require suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The suspension of CORE ALTERATIONS and movement of irradiated fuel assemblies does not preclude actions to place a fuel assembly in a safe location; the suspension of positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SHUTDOWN MARGIN is maintained.

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

These ACTIONS minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters (and to continue this action until restoration is accomplished) in order to provide the required inverter supplied Preferred AC power to the plant instrument and control systems.

The Completion Time of "immediately" is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without inverter supplied Preferred AC power.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

A description of the basis for this SR is provided in the Bases for SR 3.8.7.1. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

None

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems - Operating

BASES

BACKGROUND

The onsite Class 1E AC, DC, and Preferred AC bus electrical power distribution systems are divided into two redundant and independent electrical power distribution trains. Each electrical power distribution train is made up of several subsystems which include the safety related buses, load centers, motor control centers, and distribution panels shown in Table B 3.8.9-1.

The Class 1E 2400 V safety related buses, Bus 1C and Bus 1D, are normally powered from offsite, but can be powered from the DGs, as explained in the Background section of the Bases for LCO 3.8.1, "AC Sources - Operating.” Each 2400 V safety related bus supplies one train of the Class 1E 480 V distribution system.

The 120 V Preferred AC buses are normally powered from the inverters. The alternate power supply for the buses is a constant voltage transformer, called the Bypass Regulator. Use of the Bypass regulator is governed by LCO 3.8.7, "Inverters - Operating." The bypass regulator is powered from the non-Class 1E instrument AC bus, Y-01. The Instrument AC bus is normally powered through an automatic bus transfer switch, an instrument AC transformer, and isolation fuses. Its normal power source is MCC-1. Loss of power to MCC-1 will cause automatic transfer of the Instrument AC bus to MCC-2.

There are two independent 125 V DC electrical power distribution subsystems.

APPLICABLE SAFETY ANALYSES

A description of the Safety Analyses applicable in MODES 1, 2, 3, and 4 is provided in the Bases for LCO 3.8.1.

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

LCO

The AC, DC, and Preferred AC bus electrical power distribution subsystems are required to be OPERABLE. The required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of AC, DC, and Preferred AC bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA.

LCO (continued)

Maintaining both trains of AC, DC, and Preferred AC bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the plant design is not defeated. Therefore, a single failure within any electrical power distribution subsystem will not prevent safe shutdown of the reactor.

OPERABLE electrical power distribution subsystems require the buses, load centers, motor control centers, and distribution panels listed in Table B 3.8.9-1 to be energized to their proper voltages. In addition, tie breakers between redundant safety related AC power distribution subsystems must be open when a 2400 V source is OPERABLE for each train. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem. If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 2400 V buses from being powered from the same offsite circuit or preclude cross connecting Class 1E 480 V subsystems when 2400 V power is available for only one train.

This LCO does not address the power source for the Preferred AC buses. The Preferred AC buses are normally powered from the associated inverter. An alternate source, the Bypass Regulator, is available to supply one Preferred bus at a time, to allow maintenance on an inverter. The proper alignment of the inverter output breakers is addressed under the inverter LCOs. Therefore a Preferred AC Bus may be considered OPERABLE when powered from either the associated inverter or the Bypass Regulator as long as the voltage and frequency of the supply is correct.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that AC, DC, and Preferred AC power is available to the redundant trains and channels of safeguards equipment, instrumentation and controls required to support engineered safeguards equipment in the event of an accident or transient.

Electrical power distribution subsystem requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are addressed in LCO 3.8.10, "Distribution Systems - Shutdown."

BASES

ACTIONS

A.1

With one or more required AC buses, load centers, motor control centers, or distribution panels, except Preferred AC buses, in one train inoperable, the redundant AC electrical power distribution subsystem in the other train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because an additional failure in the power distribution systems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels must be restored to OPERABLE status within 8 hours.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combinations of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal “time zero” for beginning the Completion Time “clock.” This will result in establishing the “time zero” at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With one Preferred AC bus inoperable, the remaining OPERABLE Preferred AC buses are capable of supporting the minimum safety functions necessary to shut down the plant and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the Preferred AC bus must be restored to OPERABLE status within 8 hours by powering it from the associated inverter or from the Bypass Regulator.

BASES

ACTIONS

B.1 (continued)

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate Preferred AC power and is a feature of the original Palisades licensing basis.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single continuous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the Preferred AC distribution system. At this time, a DC bus could again become inoperable, and Preferred AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal “time zero” for beginning the Completion Time “clock.” This will result in establishing the “time zero” at the time the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With one or more DC bus in one train inoperable, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the required DC buses must be restored to OPERABLE status within 8 hours by powering the bus from the associated battery or charger.

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components which would be without power and is a feature of the original Palisades licensing basis.

ACTIONS

C.1 (continued)

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single continuous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the Preferred DC distribution system. At this time, a AC bus could again become inoperable, and Preferred AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal “time zero” for beginning the Completion Time “clock.” This will result in establishing the “time zero” at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to an operating condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

Condition E corresponds to a degradation in the electrical distribution system that causes a required safety function to be lost. When more than one Condition is entered, and this results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This surveillance verifies that the required AC, DC, and Preferred AC bus electrical power distribution subsystems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained.

For those buses which have undervoltage alarms in the control room, correct voltage may be verified by the absence of an undervoltage alarm.

For those buses which have only one possible power source and have undervoltage alarms in the control room, correct breaker alignment may be verified by the absence of an undervoltage alarm.

A Preferred AC Bus may be considered correctly aligned when powered from either the associated inverter or from the bypass regulator. A mechanical interlock prevents connecting two or more Preferred AC Buses to the Bypass Regulator. LCO 3.8.7 and LCO 3.8.8 address the condition of supplying a Preferred AC Bus from the bypass regulator.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

None

INSERT Bases 3.8.9

TABLE B 3.8.9-1 (page 1 of 1)
Required Electrical Distribution Trains

TYPE	VOLTAGE	LEFT TRAIN	RIGHT TRAIN
AC Power Distribution Subsystems	2400	Bus 1C	Bus 1D
	480	Bus 11	Bus 12
	480	Bus 19	Bus 20
	480	MCC 1	MCC 2
	480	MCC 7	MCC 8
	480	MCC 21	MCC 22
	480	MCC 23	MCC 24
	480	MCC 25	MCC 26
DC Power Distribution Subsystems	125	Bus D10-L	Bus D20-L
	125	Bus D10-R	Bus D20-R
	125	Pnl D11A	Pnl D21A
	125	Pnl D11-1	Pnl D21-1
	125	Pnl D11-2	Pnl D21-2
Preferred AC Subsystems	120	Bus Y-10	Bus Y-20
	120	Bus Y-30	Bus Y-40

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the AC, DC, and Preferred AC bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

A description of the Safety Analyses applicable during MODES 5 and 6 is provided in the Bases for LCO 3.8.2, "AC Sources - Shutdown."

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

LCO

This LCO requires those, and only those, AC, DC, and Preferred AC distribution subsystems to be OPERABLE which are necessary to support equipment required by other LCOs.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

This LCO does not address the power source for the Preferred AC buses. The Preferred AC buses are normally powered from the associated inverter. An alternate source, the Bypass Regulator, is available to supply one Preferred bus at a time, to allow maintenance on an inverter. The proper alignment of the inverter output breakers is addressed under LCO 3.8.8, "Inverters - Shutdown." Therefore a Preferred AC Bus may be considered OPERABLE when powered from either the associated inverter or the Bypass Regulator as long as the voltage and frequency of the supply is correct.

APPLICABILITY

The electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that equipment and instrumentation is available to:

a. Provide coolant inventory makeup,

b. Mitigate a fuel handling accident,

c. Mitigate shutdown events that can lead to core damage, and

APPLICABILITY (continued)

d. Monitoring and maintaining the plant in a cold shutdown condition and refueling condition.

This LCO is applicable during movement of irradiated fuel assemblies even if the plant is in a condition other than MODE 5 or 6. This LCO provides the necessary ACTIONS if the electrical power distribution subsystems required by this LCO become unavailable during movement of irradiated fuel assemblies.

The electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.9, "Distribution Systems - Operating."

ACTIONS

A.1

Since the distribution systems are only required to support features required by other LCOs, the option to declare those affected required features to be inoperable assures that appropriate ACTIONS will be implemented in accordance with the affected LCOs.

A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Required Action A.1 may involve undesired and unnecessary administrative efforts, therefore, Required Actions A.2.1 through A.2.5 provide alternate, but sufficiently conservative, actions for unplanned losses of power to distribution systems.

Required Actions A.2.1, A.2.2, A.2.3, and A.2.5 require suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions, and declaration that affected shutdown cooling trains are inoperable. The suspension of CORE ALTERATIONS and movement of irradiated fuel assemblies does not preclude actions to place a fuel assembly in a safe location; the suspension of positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SHUTDOWN MARGIN is maintained.

These ACTIONS minimize the probability or the occurrence of postulated events. It is further required (Required Action A.2.4) to immediately initiate action to restore the required distribution subsystems (and to continue this action until restoration is accomplished) in order to provide the necessary electrical power to the plant safety systems.

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5 (continued)

The Completion Time of "immediately" is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

A description of the basis for this SR is provided in the Bases for SR 3.8.9.1. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

None

INSERT Bases 3.9.1

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Primary Coolant System (PCS), and refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. The refueling operations boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The REFUELING BORON CONCENTRATION limit is defined in Section 1.1, “Definitions.” Plant procedures ensure the specified boron concentration in order to maintain the reactor core subcritical by at least 5% $\Delta\rho$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures. During evolutions where plant procedures allow manipulation of control rods or where conditions could result in inadvertent control rod withdrawal, such as reactor vessel head removal, the boron concentration must be sufficient to assure that the reactor core will remain subcritical by at least 5% $\Delta\rho$ without taking credit for the negative reactivity provided by the control rods (i.e., assuming all rods fully withdrawn). During evolutions where the control rods are inserted, plant procedures do not allow manipulation of control rods, and conditions do not exist that could result in inadvertent rod withdrawal, such as MODE 6 operations with the Upper Guide Structure in place (other than during head removal). Therefore, credit may be taken for the negative reactivity provided by the control rods when determining the boron concentration necessary to assure that the reactor core will remain subcritical by at least 5% $\Delta\rho$.

The Palisades Nuclear Plant design criteria requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) System is capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

BACKGROUND (continued)

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the PCS is cooled and depressurized the vessel head is unbolted and the head is removed. The refueling cavity is then flooded with borated water from the safety injection refueling water tank into the open reactor vessel by gravity feeding or by the use of the spent fuel cooling, safety injection pumps, or charging pumps.

The pumping action of the SDC System in the PCS and the natural circulation due to thermal driving head in the reactor vessel mix the added concentrated boric acid with the water in the refueling cavity. The SDC System is in operation during refueling (see LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level") to provide forced circulation in the PCS and to assist in maintaining the REFUELING BORON CONCENTRATION in the PCS, and the refueling cavity.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident analysis and is conservative for MODE 6. The REFUELING BORON CONCENTRATION limit is based on the core reactivity and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the reactor core will remain subcritical by at least 5% $\Delta\rho$ during the refueling operation.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling cavity, and the reactor vessel form a single connected water mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in B 3.1.1, "SHUTDOWN MARGIN."

Boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The LCO requires that a minimum boron concentration be maintained in the PCS, and refueling cavity while in MODE 6. The boron concentration limit specified ensures the reactor core will remain subcritical by at least 5% $\Delta\rho$ during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures the reactor core will remain subcritical by at least 5% $\Delta\rho$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensures that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the plant in compliance with the LCO. If the boron concentration of any coolant volume in the PCS or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position, or normal cooldown of the coolant volume for the purpose of system temperature control.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for plant conditions.

ACTIONS

A.3 (continued)

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the PCS and the refueling cavity is within the limit. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.3

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range channels (NI-1/3 and NI-2/4) are used during refueling operations to monitor the core reactivity condition. The installed source range channels are part of the Nuclear Instrumentation System. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors in place of installed detectors is permitted, provided the LCO requirements are met.

The installed source range channels utilize fission detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers five decades of neutron flux (1E+5 cps). The detectors provide continuous visual and audible indication in the control room to alert operators to a possible dilution accident. The Nuclear Instrumentation System is designed in accordance with the criteria presented in Reference 1.

If used, portable detectors should be functionally equivalent to the installed source range channels.

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range channels are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that normally available SHUTDOWN MARGIN would be reduced, but there is sufficient time for the operator to take corrective actions.

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

LCO

This LCO requires two source range channels OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each channel must provide visual indication and at least one of the two channels must provide an audible count rate function in the control room. Therefore, with no audible count rate function from at least one channel, both source range channels would be inoperable.

BASES

APPLICABILITY

In MODE 6, the source range channels must be OPERABLE to detect changes in core reactivity. There is no other direct means available to check core reactivity levels.

In MODES 3, 4, and 5, the installed source range channels are required to be OPERABLE by LCO 3.3.9, "Neutron Flux Monitoring Channels." In MODES 1, 2, and 3, one source range channel is required by LCO 3.3.8, “Alternate Shutdown System.”

ACTIONS

A.1 and A.2

With only one source range channel OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.

B.1

With no source range channel OPERABLE, action to restore a channel to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until one source range channel is restored to OPERABLE status.

B.2

With no source range channel OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range channel are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to verify that the required boron concentration exists. The Completion Time of once per 12 hours is sufficient to obtain and analyze a primary coolant and refueling cavity sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this period.

BASES

ACTIONS

B.2 (continued)

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 3.9.2 B.2 must be initially performed within 12 hours without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions, but does not require the two source range channels to have the same reading. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.9.2.2

The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 7.6

2. FSAR, Section 14.3

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, a release of fission product radioactivity within the containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are filtered, closed or capable of being closed. Since there is no potential for containment pressurization, the 10 CFR 50, Appendix J leakage criteria and tests are not required. In MODE 5, no accidents are assumed which will result in a release of radioactive material to the containment atmosphere. Therefore, no requirements are stipulated for containment penetrations in MODE 5.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within applicable 10 CFR 50.67 limits. Additionally, the containment structure provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment with the equipment hatch closed, the hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment with the equipment hatch removed, the OPERABILITY requirements of the Fuel Handling Area Ventilation System must be met. These OPERABILITY requirements are provided in LCO 3.7.12, "Fuel Handling Area Ventilation System."

BACKGROUND (continued)

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed. An exception, however, is provided for the personnel air lock. It is acceptable to have both doors of the personnel air lock open simultaneously provided the equipment hatch is open.

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Vent System includes a 12 inch purge penetration and two 8 inch exhaust penetrations. During MODES 1, 2, 3, and 4, the valves in the purge and vent penetrations are secured in the closed position and venting the containment is accomplished using the Clean Waste Receiving Tank (CWRT) vent line. The two valves in the CWRT vent line penetration are closed automatically by a Containment High Radiation signal. Neither the Containment Purge and Vent System, nor the CWRT vent line is subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The Purge and Vent System is used for this purpose. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment with either the Containment Purge and Vent System in operation, or the CWRT aligned for containment venting, the associated isolation valves must be capable of being closed by an OPERABLE channel of radiation instrumentation required by LCO 3.3.6, "Refueling Containment High Radiation Instrumentation."

BACKGROUND (continued)	Other containment penetrations that provide direct access from containment atmosphere to outside atmosphere that are not capable of being closed by an OPERABLE Refueling Containment High Radiation signal must be isolated on at least one side. Containment penetrations “that provide direct access from containment atmosphere to outside atmosphere” are those which would allow passage of air containing radioactive particulates to migrate from inside the containment to the atmosphere outside the containment even though no measurable differential pressure existed. Specifically, they do not include penetrations which are filtered, or penetrations whose piping is filled with liquid. Isolation may be achieved by a manual or automatic isolation valve, blind flange, or equivalent. Equivalent isolation methods, authorized under the provisions of 10 CFR 50.59, may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements.
APPLICABLE SAFETY ANALYSES	During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). The requirements of LCO 3.9.6, "Refueling Cavity Water Level," (and the minimum decay time of 48 hours required by the Operating Requirements Manual) prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are less than applicable 10 CFR 50.67 limits. Containment penetration isolation is not required by the fuel handling accident to maintain offsite doses within applicable 10 CFR 50.67 limits, but operating experience indicates that containment isolation provides significant reduction of the resulting offsite doses. Therefore, the Containment Penetrations satisfy the requirements of Criterion 4 of 10 CFR 50.36(c)(2).
LCO	This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires the equipment hatch, air locks and any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment penetrations.

LCO (continued)

For the OPERABLE containment penetrations, this LCO ensures that these penetrations are isolable by the Refueling Containment High Radiation instrumentation. The OPERABILITY requirements for this LCO do not assume a specific closure time for the valves in these penetrations since the accident analysis makes no specific assumptions about containment closure time after a fuel handling accident.

LCO 3.9.3.a is modified by a Note which allows the equipment hatch to be opened if the Fuel Handling Area Ventilation System is in compliance with LCO 3.7.12. LCO 3.9.3.b is modified by a Note which allows both doors of the personnel air lock to be simultaneously opened provided the equipment hatch is opened. In the event of a fuel handling accident inside containment with both doors in the personnel air lock open and the equipment hatch open, the Fuel Handling Area Ventilation System would be available to filter the fission products in the containment atmosphere prior to their being released to the environment and thereby significantly reducing the offsite dose.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment."

In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the Refueling Containment High Radiation instrumentation not being capable of automatic actuation when the purge and exhaust valves are open, the plant must be placed in a condition in which containment closure is not needed.

ACTIONS (continued)

A.1 and A.2 (continued)

This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the valves in unisolated penetrations which provide a direct path from the containment atmosphere to the outside atmosphere will demonstrate that the valves are not blocked from closing. Also, the Surveillance will demonstrate that each valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE Refueling Containment High Radiation signal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.9.3.2

This Surveillance demonstrates that each automatic isolation valve providing direct access from the containment atmosphere to the outside atmosphere valve actuates to its isolation position on an actual or simulated high radiation signal.

The SR is modified by a Note which requires only the valves in unisolated penetrations to be tested. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 14.19

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).

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 $\leq$ 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Sections 6.1 and 14.3

INSERT Bases 3.9.4

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Sections 6.1 and 14.3

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Cavity Water Level

BASES

BACKGROUND	The performance of CORE ALTERATION or the movement of irradiated fuel assemblies within containment requires a minimum water level greater than or equal to the 647 ft elevation. During refueling this maintains sufficient water level in the refueling cavity and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to less than applicable 10 CFR 50.67 limits.
APPLICABLE SAFETY ANALYSES	During core alterations and during movement of irradiated fuel assemblies, the water level in the refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide (RG) 1.183 (Ref. 1). The fuel handling accident analysis inside containment is described in Reference 2. A minimum water level of 647 feet provides 22.5 feet of water above the damaged fuel. RG 1.183 specifies a method to address the condition of less than 23 feet of overlying water above the damaged fuel.

BASES

APPLICABLE SAFETY ANALYSIS (continued)	The analyses in Reference 2 demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained within applicable 10 CFR 50.67 limits. Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2).
LCO	A minimum refueling cavity water level greater than or equal to the 647 ft elevation is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are less than applicable 10 CFR 50.67 limits.
APPLICABILITY	LCO 3.9.6 is applicable during CORE ALTERATIONS, and when moving fuel assemblies in the presence of irradiated fuel assemblies in containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Spent Fuel Pool Water Level."
ACTIONS	<u>A.1 and A.2</u> With a water level below the 647 ft elevation, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level corresponding to the 647 ft elevation ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required elevation limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Regulatory Guide 1.183

2. FSAR, Section 14.19