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PNP 2016-021

April 14, 2016

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

SUBJECT: Submittal of Changes to Palisades Nuclear Plant Technical Specifications Bases

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

REFERENCES: 1. Entergy Nuclear Operations Inc. letter, PNP 2014-088, "Report to NRC Changes to Technical Specifications Bases," dated September 15, 2014 (ADAMS Accession No. ML14260A013)

Dear Sir or Madam:

In accordance with Palisades Nuclear Plant (PNP) Technical Specification Section, 5.5.12d, "Technical Specifications (TS) Bases Control Program," which requires that changes to the TS Bases, implemented without prior Nuclear Regulatory Commission (NRC) approval, be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Entergy Nuclear Operations, Inc. hereby submits a listing of all PNP TS bases changes since issuance of the previous TS Bases submittal, dated September 15, 2014 (Reference 1), and identifies the affected sections and describes the nature of the change in Attachment 1. Attachment 2 provides page change instructions, and copies of the revised TS Bases Title Page, the List of Effective Pages, and the TS Bases sections listed in Attachment 1.

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

Sincerely,

A handwritten signature in black ink, appearing to read "JAH".

JAH/jse

Attachments: 1. List of Palisades Nuclear Plant Technical Specifications Bases Changes
2. Revised Technical Specifications Bases

ADD
NR R

PNP 2016-021
Page 2 of 2

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC

ATTACHMENT 1

**LIST OF PALISADES NUCLEAR PLANT
TECHNICAL SPECIFICATIONS BASES CHANGES**

Date	Affected Bases	Change Description
December 3, 2014	Section B 3.0	<p>Bases revised to reflect license amendment 252, dated October 22, 2014 (ADAMS Accession No. ML14283A287), which adopted Technical Specification Task Force (TSTF) traveler TSTF-427, "Allowance for Non-Technical Specification Barrier Degradation on Supported System OPERABILITY."</p> <p>Under this amendment, Limiting Condition for Operation (LCO) 3.0.9, associated with the unavailability of barriers, was added to the Technical Specifications.</p>
July 29, 2015	Sections B 3.4.9, B 3.7.10, and B 3.7.11	<p>Bases revised to reflect license amendment 256, dated May 18, 2015 (ADAMS Accession No. ML15103A059), which adopted Technical Specification Task Force (TSTF) traveler TSTF-426, "Revise or Add Actions to Preclude Entry into LCO 3.0.3 – RITSTF Initiatives 6b and 6c."</p> <p>Under this amendment, entry into LCO 3.0.3 was precluded for the pressurizer heaters, control room filtration, and control room cooling.</p>

<p>April 14, 2016</p>	<p>Sections B 2.1.2, B 3.1.1, B 3.3.1, B 3.3.2, B 3.3.6, B 3.3.10, B 3.4.13, B 3.4.16, B 3.4.17, B 3.7.2, B 3.7.12, B 3.7.13, B 3.7.14, B 3.7.17, B 3.9.3, and B 3.9.6</p>	<p>Bases revised to reflect license amendment 226, dated September 28, 2007 (ADAMS Accession No. ML072470676) and May 7, 2008 (ADAMS Accession No. ML15334A412). The license amendment approved performing radiological consequence analysis using alternative source terms at Palisades. The Bases were revised to incorporate references to 10 CFR 50.67, "Accident source term," and to update other related information.</p> <p>Entergy Nuclear Operations, Inc. Condition Report CR-PLP-2016-01462 was initiated to document that the Bases change associated with this license amendment was not updated in accordance with the timeliness requirements of 10 CFR 50.71(e).</p>
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ATTACHMENT 2

REVISED TECHNICAL SPECIFICATIONS BASES

Page Change Instructions

Title Page

List of Effective Pages

Technical Specification Bases Sections

B 2.1.2

B 3.0

B 3.1.1

B 3.3.1

B 3.3.2

B 3.3.6

B 3.3.10

B 3.4.9

B 3.4.13

B 3.4.16

B 3.4.17

B 3.7.2

B 3.7.10

B 3.7.11

B 3.7.12

B 3.7.13

B 3.7.14

B 3.7.17

B 3.9.3

B 3.9.6

162 Pages Follow

Technical Specifications Bases

Page Change Instructions

Revise your copy of the Palisades Nuclear Plant Technical Specifications Bases by removing the pages identified below and inserting the revised pages. Vertical lines in the margin indicate the area of change.

LIST TECHNICAL SPECIFICATION BASES PAGES	
REMOVE	INSERT
– Technical Specification Bases Title Page Revised 09/15/14 (1 page)	– Technical Specification Bases Title Page Revised 04/14/2016 (1 page)
– Palisades Tech Spec Bases List of Effective Pages Revised 09/15/14 (3 pages)	– Palisades Tech Spec Bases List of Effective Pages Revised 04/14/2016 (3 pages)
– Pages B 2.1.2-1 - B 2.1.2-4 Amendment 189 (4 pages)	– Pages B 2.1.2-1 - B 2.1.2-4 Amendment 226 (4 pages) Revised 04/14/2016
– B 3.0-1 – B 3.0-18 Amendment 251 (18 pages) Revised 06/13/13	– B 3.0-1 – B 3.0-21 Amendment 252 (21 pages) Revised 12/03/2014
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– Pages B 3.3.1-1 - B 3.3.1-35 Revised 08/30/2011 (35 pages)	– Pages B 3.3.1-1 - B 3.3.1-35 Amendment 226 (35 pages) Revised 04/14/2016
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- Pages B 3.4.16-1 - B 3.4.16-5 Revised 02/24/2005 (5 pages)	- Pages B 3.4.16-1 - B 3.4.16-5 Amendment 226 (5 pages) Revised 04/14/2016
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PALISADES PLANT
FACILITY OPERATING LICENSE DPR-20
APPENDIX A

TECHNICAL SPECIFICATIONS

BASES

PALISADES TECHNICAL SPECIFICATIONS BASES
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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).

BASES

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.

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.

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

BASES

LCO	LCO 3.0.1 through LCO 3.0.9 establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
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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 plant is in the MODES or other specified conditions of the Applicability statement of each Specification).
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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:
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- a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and
- b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

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.)

BASES

LCO 3.0.2
(continued)

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.

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 reach 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.

BASES

LCO 3.0.3
(continued)

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. A Condition exists for which the Required Actions have now been performed.
- c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the 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 reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 29 hours, because the total time for reaching 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 reach 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."

BASES

LCO 3.0.3
(continued)

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.

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 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 permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required 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 in accordance with the provisions of the Required Actions.

LCO 3.0.4.b allows 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, 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.

BASES

LCO 3.0.4
(continued)

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.

BASES

LCO 3.0.4
(continued)

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 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.

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.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition 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.

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.

BASES

LCO 3.0.6
(continued)

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.

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.

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.

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.

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

BASES

LCO 3.0.8 (continued) 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.

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.

BASES

LCO 3.0.9 (continued) 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 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.

BASES

LCO 3.0.9 (continued) 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).

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 3.0.1 SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other 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 the OPERABILITY of systems and components, and 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. 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.

Surveillances do not have to be performed when the plant is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. 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.

BASES

SR 3.0.1
(continued)

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

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

BASES

SR 3.0.2
(continued)

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. 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. An example of where SR 3.0.2 does not apply is the Containment Leak Rate Testing Program.

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.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

BASES

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 completed 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.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of 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 unit conditions, 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.) 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.

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 as an operational convenience to extend Surveillance intervals.

BASES

SR 3.0.3
(continued)

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 plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. 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. 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.

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.

BASES

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified Condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the plant.

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.

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 system, subsystem, division, component, device, or variable is inoperable or 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 inoperable equipment. When equipment is inoperable, 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 MODES or other 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 MODE or other specified condition changes.

SR 3.0.4 does not restrict changing MODES or other 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.

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.

BASES

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 MODE or other 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

ACTIONSA.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% Δp 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 $\Delta p/ppm$ is assumed, this combination of parameters will increase the SDM by 1% Δp . 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 Frequency of 24 hours for the verification of SDM is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which may include performing a boron concentration analysis, and completing the calculation.

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 3.3 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.

BASES

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.

BASES

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, $P_{trip} + \Delta P$.

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, $\leq 30\%$. 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 $\leq 15\%$ 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. High Pressurizer Pressure Trip

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. Thermal Margin/Low Pressure (TM/LP) Trip

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. Loss of Load Trip

The LCO requires four Loss of Load Trip Function channels to be OPERABLE in MODE 1 with THERMAL POWER \geq 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. Containment High Pressure Trip

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 \geq 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 \geq 17% RTP. For this Completion Time, "following a shutdown" means this Required Action does not have to be completed until prior to THERMAL POWER \geq 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.

BASES

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

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 12 hour Frequency is reasonable based on engineering judgment and plant operating experience.

SR 3.3.1.3

A daily calibration (heat balance) is performed when THERMAL POWER is $\geq 15\%$. The daily calibration consists of adjusting the "nuclear power calibrate" potentiometers to agree with the calorimetric calculation if the absolute difference is $\geq 1.5\%$. 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 daily calibration (heat balance) procedure. Performance of the daily 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 Frequency of 24 hours is based on plant operating experience and takes into account indications and alarms located in the control room to detect deviations in channel outputs.

The Frequency is modified by a Note indicating this Surveillance must be performed within 12 hours after THERMAL POWER is $\geq 15\%$ 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	Group 4
	<u>Rods $\geq 128''$ withdrawn</u>	<u>Rods $< 128''$ withdrawn</u>
$\leq 100\%$	$-0.020 \leq (AO-ASI) \leq 0.020$	$-0.040 \leq (AO-ASI) \leq 0.040$
< 95	$-0.033 \leq (AO-ASI) \leq 0.020$	$-0.053 \leq (AO-ASI) \leq 0.040$
< 90	$-0.046 \leq (AO-ASI) \leq 0.020$	$-0.066 \leq (AO-ASI) \leq 0.040$
< 85	$-0.060 \leq (AO-ASI) \leq 0.020$	$-0.080 \leq (AO-ASI) \leq 0.040$
< 80	$-0.120 \leq (AO-ASI) \leq 0.080$	$-0.140 \leq (AO-ASI) \leq 0.100$
< 75	$-0.120 \leq (AO-ASI) \leq 0.080$	$-0.140 \leq (AO-ASI) \leq 0.100$
< 70	$-0.120 \leq (AO-ASI) \leq 0.080$	$-0.140 \leq (AO-ASI) \leq 0.100$
< 65	$-0.120 \leq (AO-ASI) \leq 0.080$	$-0.140 \leq (AO-ASI) \leq 0.100$
< 60	$-0.160 \leq (AO-ASI) \leq 0.120$	$-0.180 \leq (AO-ASI) \leq 0.140$
< 55	$-0.160 \leq (AO-ASI) \leq 0.120$	$-0.180 \leq (AO-ASI) \leq 0.140$
< 50	$-0.160 \leq (AO-ASI) \leq 0.120$	$-0.180 \leq (AO-ASI) \leq 0.140$
< 45	$-0.160 \leq (AO-ASI) \leq 0.120$	$-0.180 \leq (AO-ASI) \leq 0.140$
< 40	$-0.160 \leq (AO-ASI) \leq 0.120$	$-0.180 \leq (AO-ASI) \leq 0.140$
< 35	$-0.160 \leq (AO-ASI) \leq 0.120$	$-0.180 \leq (AO-ASI) \leq 0.140$
< 30	$-0.160 \leq (AO-ASI) \leq 0.120$	$-0.180 \leq (AO-ASI) \leq 0.140$
< 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 \geq 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 31 day Frequency is adequate, based on operating experience of the excore linear amplifiers and the slow burnup of the detectors. The excore readings are a strong function of the power produced in the peripheral fuel bundles and do not represent an integrated reading across the core. Slow changes in neutron flux during the fuel cycle can also be detected at this Frequency.

SR 3.3.1.5

A CHANNEL FUNCTIONAL TEST is performed on each RPS instrument channel, except Loss of Load and High Startup Rate, every 92 days 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 Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 5).

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.6

A calibration check of the power range excore channels using the internal test circuitry is required every 92 days. 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 Frequency of 92 days is acceptable, based on plant operating experience, and takes into account indications and alarms available to the operator in the control room.

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 at least once per refueling interval with applicable extensions.

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 is the performance of a CHANNEL CALIBRATION every 18 months.

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

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 daily calorimetric calibration (SR 3.3.1.3) and the monthly 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
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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 Δ T 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 Δ T 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 Δ T 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 Δ T 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)	3.3.1 (#6 & 7) & 3.3.3 (#2a, 2b, 7b, 7c)
PT-0751A and PT-0752A Pressure Signal (C-150/150A)	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 Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 5).

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

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

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 at least once per refueling interval with applicable extensions.

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

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 Frequency of 18 months is based on plant operating experience with regard to channel OPERABILITY, and is consistent with the testing of other manually actuated functions.

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 Frequency is based upon the assumption of an 18 month calibration interval in the setpoint determination.

REFERENCES

1. FSAR, Section 7.3
 2. FSAR, Section 14.19
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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 $\leq 2.2E+5$ cpm. This setpoint is high enough to avoid inadvertent actuation in the event of normal background radiation fluctuations during testing, but low enough to isolate the ES pump room in the event of radiation levels indicative of a LOCA and excessive leakage during recirculation of primary coolant through the ES pump room.

BASES

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

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

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

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

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

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

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.10.2

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

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

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

SR 3.3.10.3

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

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

REFERENCES

1. FSAR, Section 7.4.5.2
 2. FSAR, Section 14.22
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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 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the capacity of the associated pressurizer heaters are verified to be ≥ 375 kW. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Frequency of 18 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

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 Frequency of 18 months is based on engineering judgement and is considered acceptable when considering the design reliability of the equipment (the jumper cable is left permanently in place and dedicated to providing the emergency feed function only), and administrative control which govern configuration management and changes to plant procedures.

REFERENCES

1. FSAR, Chapter 14
 2. FSAR, Section 4.3.7
 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.
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B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.13 PCS Operational LEAKAGE

BASES

BACKGROUND

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

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

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

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

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

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

BASES

BACKGROUND
(continued)

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

- (1) Part of the primary coolant system, or
- (2) Connected to the primary coolant system, up to and including any and all of the following:
 - (i) The outermost containment isolation valve in system piping which penetrates the containment,
 - (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate the containment,
 - (iii) The pressurizer safety valves and PORVs.

APPLICABLE
SAFETY ANALYSES

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

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

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

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

The CRE (Ref 4 and 5) accident with primary fluid release through the

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

BASES

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.

BASES

ACTIONS
(continued)

B.1 and B.2

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

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

**SURVEILLANCE
REQUIREMENTS**

SR 3.4.13.1

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

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

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

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

BASES

SURVEILLANCE
REQUIREMENTS
(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.

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

SR 3.4.13.2

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

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

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

BASES (continued)

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

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 $\mu\text{Ci/gm}$ 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 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131, and the gross specific activity in the primary coolant is limited to the number of $\mu\text{Ci/gm}$ 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 $\geq 500^{\circ}\text{F}$, 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 $< 500^{\circ}\text{F}$, 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 $40 \mu\text{Ci/gm}$ is not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample.

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

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

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

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

BASES

ACTIONS
(continued)

B.1

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

The change within 6 hours to MODE 3 with PCS average temperature < 500°F lowers the saturation pressure of the primary coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is required to reach MODE 3 below 500°F from full power conditions and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.16.1

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

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

BASES

**SURVEILLANCE
REQUIREMENTS**
(continued)

SR 3.4.16.2

This Surveillance is performed to ensure iodine remains within limits during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross activity is monitored every 7 days. The Frequency, between 2 hours and 6 hours after any power change of $\geq 15\%$ 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 $\geq 15\%$ 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 every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The \bar{E} determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for \bar{E} is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes \bar{E} does not change rapidly.

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

REFERENCES

1. FSAR, Section 14.15
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B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

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

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

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

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

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

BASES

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.

BASES

LCO
(continued)

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

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

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

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

BASES

LCO
(continued)

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

APPLICABILITY

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

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

ACTIONS

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

A.1 and A.2

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

BASES

ACTIONS
(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.

BASES

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.

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
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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.

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 Frequency for this SR is every 18 months. This 18 month Frequency demonstrates the valve closure time at least once per refueling cycle. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

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 3.7.10 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

BASES

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).

BASES

**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.

BASES

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

BASES

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

BASES

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 once every month provides an adequate check on this system.

Monthly 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 31 day Frequency is based on the known reliability of the equipment, and the two train redundancy available.

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 Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.

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).
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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.

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. An 18 month Frequency is appropriate, since significant degradation of the CRV Cooling is slow and is not expected over this time period.

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.
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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.

BASES

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 ≥ 5840 cfm and ≤ 8760 cfm. The Frequency of 18 months is consistent with the test for filter performance and other filtration SRs.

BASES

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 31 day Frequency is based on operating experience which has shown that these components usually pass the SR when tested at this Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.67
 2. FSAR, Section 7.4.5.2
 3. FSAR, Section 14.22
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B 3.7 PLANT 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, or for a fuel handling accident inside containment, 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).

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 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.

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.

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.

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 daily 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
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B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube leakage 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 $\mu\text{Ci}/\text{gm}$. 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 $\mu\text{Ci}/\text{gm}$ 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 $\leq 0.10 \mu\text{Ci/gm DOSE}$ 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 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 50.67
 2. FSAR, Section 14.14
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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."

BASES

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."

BASES

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.

BASES

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.

BASES

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 is performed every 7 days during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance provides assurance that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in an excessive release of fission product radioactivity to the environment.

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.

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.9.3.2 (continued)

The SR is modified by a Note which requires only the valves in unisolated penetrations to be tested. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. LCO 3.3.6, "Refueling Containment High Radiation Instrumentation," requires a CHANNEL CHECK every 7 days, a CHANNEL FUNCTIONAL TEST every 31 days and a CHANNEL CALIBRATION every 18 months to ensure the channel OPERABILITY during refueling operations. These surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. FSAR, Section 14.19
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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

A.1 and A.2

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 Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.183
 2. FSAR, Section 14.19
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