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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

August 21, 2006

L-MT-06-054
10 CFR 50.90
10 CFR 50.67

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

Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Supplemental Submittal Regarding Full Scope Application of an Alternative Source Term (AST)(TAC No. MC8971)

References: 1) NMC letter to NRC, "License Amendment Request: Full Scope Application of an Alternative Source Term," (ADAMS Accession No. ML052640366) dated September 15, 2005.

2) NRC letter to NMC, "Monticello Nuclear Generating Plant (MNGP) - Issuance of Amendment for the Conversion to the Improved Technical Specifications with Beyond-Scope Issues," (ADAMS Accession No. ML061070577) dated June 5, 2006.

Pursuant to 10 CFR 50.67 and 10 CFR 50.90, the Nuclear Management Company, LLC, (NMC) hereby provides the Alternative Source Term Technical Specifications (TS) changes in Improved Technical Specifications (ITS) format for the Monticello Nuclear Generating Plant (MNGP), Operating License DPR-22.

The MNGP License Amendment Request (LAR) for Full Scope Implementation of an Alternative Source Term (AST) (Reference 1) contained proposed TS changes in the MNGP Custom TS format. Subsequent to the submittal of Reference 1, the NRC staff approved the MNGP Conversion to ITS in Reference 2. This submittal provides proposed TS changes in the ITS format to reflect the AST LAR (Reference 1) currently being reviewed by the NRC staff. It also includes additional proposed TS changes to implement those portions of Technical Specifications Task Force (TSTF)-51, Revision 2 that were not applicable in MNGP Custom TS and to provide additional changes to the Control Room Emergency Filtration Instrumentation and Control Room Ventilation TS.

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Enclosure 1 provides the introduction, summary, background, description of the proposed changes, and technical and regulatory evaluations. The no significant hazards and environmental considerations determinations submitted in Reference 1 were reviewed and determined to be applicable and continue to bound the proposed changes discussed herein. Enclosure 2 provides a mark-up of the proposed conversion changes from the MNGP Custom TS (CTS) to the MNGP ITS. Enclosure 3 provides a mark-up of the other proposed ITS changes. Enclosure 4 provides a mark-up of the proposed MNGP ITS Bases changes (for information only). Enclosure 5 provides a retyped version of the conversion from MNGP CTS to ITS. Enclosure 6 provides a retyped version of the other proposed ITS changes.

This letter contains no new commitments.

Nuclear Management Company, LLC as stated in the previous License Amendment Request (Reference 1), still requests approval of the proposed amendment by January 4, 2007, to support the currently scheduled March 2007 MNGP Refueling Outage.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 21, 2006.

John T. Conway

John T. Conway
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (6)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

Submittal of Improved Technical Specification and Bases Pages for the Monticello Nuclear Generating Plant Regarding Full Scope Application of an Alternative Source Term (AST)

Introduction

Pursuant to 10 CFR 50.67 and 10 CFR 50.90, the Nuclear Management Company, LLC, (NMC) hereby requests changes to the Technical Specifications (TS) for the Monticello Nuclear Generating Plant (MNGP), Operating License No. DPR-22. This License Amendment Request (LAR) proposes changes to the MNGP TS to reflect a previous NMC LAR submittal (Reference 1) (ADAMS Accession No. ML052640366) in Improved Technical Specifications (ITS) and ITS Bases page format. The MNGP Conversion from Custom TS to ITS was approved by the NRC by letter dated June 5, 2006 (Reference 2) (ADAMS Accession No. ML061070577). This request also includes additional ITS changes to implement those portions of Technical Specifications Task Force (TSTF)-51, Revision 2, that were not applicable to the MNGP Custom TS.

By letter dated April 29, 2004, NMC requested a LAR for MNGP regarding a Selective Scope Application of an Alternative Source Term for Re-Evaluation of a Fuel Handling Accident (Reference 4) (ADAMS Accession No. ML041450022). The changes that were requested by the Reference 4 submittal were subsequently included in the June 29, 2005, NMC submittal for the MNGP Conversion to Improved Technical Specifications.

NMC's Selective Scope AST - FHA LAR (Reference 4) included the majority of the Technical Specification Task Force (TSTF)-51, Revision 2 changes; however, not all TSTF-51, Revision 2, changes could be implemented at MNGP at that time. Implementation of the remaining TSTF-51, Revision 2 changes could only be implemented based upon the re-evaluation of the Fuel Handling Accident (FHA) performed for Full Scope Implementation of an Alternative Source Term (AST) at MNGP and the conversion to ITS. This submittal includes the portion of the TSTF-51, Revision 2 changes which required conversion to ITS prior to adoption at MNGP.

10 CFR 50.67, "Accident source term," provides a mechanism for currently licensed nuclear power reactors to replace the traditional source term used in design basis accident analyses with an alternative source term. Under this provision, licensees who seek to revise the accident source term in design basis radiological consequence analyses must apply for a license amendment under 10 CFR 50.90.

This enclosure provides the introduction, background, description of the proposed changes, and technical and regulatory evaluations.

Background

By letter dated June 29, 2005, NMC submitted a LAR converting MNGP Custom Technical Specifications (CTS) to Improved Technical Specifications (ITS) (ADAMS Accession No. ML051960175). That submittal reflected the previously submitted NMC Selective Scope AST - FHA LAR (Reference 4).

By letter dated September 15, 2005, NMC submitted a License Amendment Request (LAR) for MNGP regarding Full Scope Application of an Alternative Source Term (Reference 1) (ADAMS Accession No. ML052640366). As part of the submittal, NMC stated that upon approval of the ITS LAR, NMC would provide a supplemental submittal updating the AST LAR with revised marked-up and retyped ITS pages.

By letter dated June 5, 2006, the NRC approved the ITS LAR for MNGP (Reference 2) (ADAMS Accession No. ML061070577). Therefore, NMC is hereby providing this submittal updating the AST LAR with revised marked-up ITS (Enclosure 2) and ITS Bases pages (Enclosure 4). NMC is also providing a supplemental submittal of additional MNGP ITS changes to implement the remainder of the TSTF-51, Revision 2, changes (Enclosure 3). Most of the TSTF-51, Revision 2 changes were previously approved by the NRC Staff for MNGP. The remainder of these changes required the support of analyses contained in the Full Scope Application of the Alternative Source Term (Reference 1) and approval of the ITS for MNGP prior to incorporation.

Description of the Proposed Changes

The following changes to the MNGP ITS (Enclosure 3) are being proposed to incorporate that portion of the TSTF-51, Revision 2, items that were not included in the Reference 1 submittal. The proposed changes are applicable to new specifications added to the MNGP TS during the ITS Conversion (Reference 2). This submittal also provides additional changes to the ITS of the wording associated with the Control Room Emergency Filtration (CREF) System Instrumentation TS and the Control Room Ventilation TS (Enclosure 3). Although the intent of the revision is the same as that proposed by the full scope AST submittal, the wording of the ITS and the ITS Bases have been rewritten. The proposed changes are consistent with the Full Scope Application of the Alternative Source Term currently under review by the NRC staff. (Reference 1)

Summary of TSTF-51, Revision 2 Related Technical Specification Changes

In addition to the Technical Specification changes associated with implementing the Full Scope AST (Reference 1), this license amendment application includes additional changes that incorporate those portions of the previously reviewed and approved generic BWR Technical Specification changes contained in TSTF-51, Revision 2, that were not applicable to the MNGP Custom TS.

TSTF-51 modifies TS requirements relating to core alterations and the handling of irradiated fuel in the secondary containment based on the recognition that after reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. Based on the AST analyses of Reference 1, the proposed license amendments remove TS requirements for certain plant systems and equipment to be operable after sufficient radioactive decay has occurred to ensure offsite dose limits are not exceeded.

Proposed changes to reflect the remaining TSTF-51 related changes involve the following ITS and ITS Bases:

Technical Specification	Affected Page(s) / Description
3.8.2 AC Sources – Shutdown	Affected Page(s): 3.8.2-1 and 3.8.2-2 (and Bases pages B 3.8.2-1 through B 3.8.2-4) Technical Specification: The Applicability is being revised to require applicability during movement of recently irradiated fuel assemblies in the secondary containment. Revise Required Actions A.2.2 and B.2 to conform to those changes. Bases: Incorporate applicability during movement of recently irradiated fuel assemblies, and describe the actual decay period required for recently irradiated fuel.
3.8.5 DC Sources - Shutdown	Affected Page(s): 3.8.5-1 (and Bases pages B 3.8.5-1 through B 3.8.5-3) Technical Specification: The Applicability is being revised to require applicability during movement of recently irradiated fuel assemblies in the secondary containment. Revise Required Action A.2 to conform to those changes. Bases: Incorporate applicability during movement of recently irradiated fuel assemblies, and describe the actual decay period required for recently irradiated fuel.
3.8.8 Distribution System - Shutdown	Affected Page(s): 3.8.8-1 (and Bases pages B 3.8.8-1 through B 3.8.8-3).

Technical Specification: The Applicability is being revised to require applicability during movement of recently irradiated fuel assemblies in the secondary containment. Required Action A.2.2 is being revised to conform to this change.

Bases: Incorporate applicability during movement of recently irradiated fuel assemblies, and describe the actual decay period required for recently irradiated fuel.

For the TS that are being revised to incorporate the remainder of the TSTF-51, Revision 2 changes, the Applicability statements are being changed to include operations involving the movement of "recently irradiated fuel assemblies." The wording of both the Conditions and the Required Actions for these TS is being modified consistent with the Applicability statement changes. The "recently irradiated fuel assemblies" terminology refers to irradiated fuel that has occupied part of a critical reactor core within the previous 24 hours. The new terminology is being used to define the conditions where fuel handling activities can involve situations for which significant radioactive releases can be postulated. Thus, the new terminology is being used to modify the operability requirements for the identified safety systems. The AST analyses demonstrate that a 24- hour decay period is sufficient to ensure secondary containment automatic isolation and CREF system automatic initiation are not required to mitigate a FHA. The 24-hour decay period will be included in the Bases for each of the modified TS.

The Applicability statements related to operations with a potential for draining the reactor vessel are unaffected by the proposed changes.

Summary of an Additional AST Technical Specification Change

As stated above this submittal provides the TS changes associated with implementing the Full Scope AST (Reference 1) in the recently approved ITS and ITS Bases page format which also includes changes of the wording associated with the Control Room Emergency Filtration (CREF) System Instrumentation TS and the Control Room Ventilation TS. Although the intent of the revision is the same as that proposed by the full scope AST submittal the TS and TS Bases wording and format have been changed. Proposed changes involving the MNGP Control Room Emergency Filtration (CREF) System Instrumentation TS and the Control Room Ventilation TS are as follows:

Technical Specification	Affected Page(s) / Description
3.3.7.1 Control Room Emergency Filtration (CREF) System Instrumentation	Affected Page(s): 3.3.7.1 through 3.3.7-3 (and Bases pages B 3.3.7.1-1 through B 3.3.7.1-8)

Technical Specification: The LCO is being revised to refer to the Functions listed in the new TS Table 3.3.7.1-1. The Applicability is being revised to refer to TS Table 3.3.7.1-1. New Required Actions A.1 and A.2 are being proposed with associated Completion Times. New Condition B is being proposed. Three new Surveillance Requirements (SRs) are being proposed for the instrumentation in the new TS Table. The new TS Table 3.3.7.1-1 is being proposed to add two additional CREF System instrumentation initiation signals, one from Reactor Building Ventilation Exhaust - High, and the other from Refueling Floor Radiation - High, that will initiate the CREF System.

Bases: The ITS Bases is being rewritten to align the wording and format of the proposed ITS changes by incorporating those proposed changes into the Bases.

3.7.5 Control Room Ventilation System	<u>Affected Page(s):</u> 3.7.5-1 and 3.7.5-2 (and Bases pages B 3.7.5-2 through B 3.7.5-4)
	<u>Technical Specification:</u> The Applicability is being revised to require applicability during movement of recently irradiated fuel assemblies in the secondary containment. Revise Conditions C and E and Required Actions C.2.1 and E.1 to conform to those changes.
	<u>Bases:</u> Incorporate applicability during movement of recently irradiated fuel assemblies, and describe the actual decay period required for recently irradiated fuel.

Technical Evaluation

Additional TSTF-51 Changes

Additional changes to the MNGP ITS, are being proposed by NMC to incorporate the remainder of the TSTF-51, Revision 2 changes (Enclosure 3). These changes were not provided with the Reference 1 submittal, because the MNGP TS, at that time, did not contain these TS requirements. A partial implementation of the TSTF-51, Revision 2, changes was accomplished for the MNGP TS with the issuance of License Amendment 145, "Alternate Source Term – Fuel Handling Accident" (Reference 3) (ADAMS Accession No. ML060600572). These changes were incorporated into the June 29, 2005, submittal for the Conversion to Improved Technical Specifications submittal to preclude the necessity for an additional License Amendment change request.

The write-up that submitted the TSTF-51, Revision 2, generic BWR/4 changes to the NRC Staff for review provides the clearest description and technical evaluation for the proposed change:

> The Technical Specification requirements for ESF features (e.g., primary/secondary containment, standby gas treatment, isolation capability) to be OPERABLE after sufficient radioactive decay has occurred to ensure off-site doses remain below the SRP limits (a small fraction of 10 CFR 50.67) may be removed. Fuel movement could still proceed prior to the amount of decay occurring but only with the appropriate ESF systems OPERABLE.
>
> Associated with this change is the deletion of OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features. This change will allow plants the flexibility to move personnel and equipment and perform work which would affect containment OPERABILITY during the handling of irradiated fuel.
>
> Following reactor shutdown, decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed changes are based on performing analyses assuming a longer decay period to take advantage of the reduced radionuclide inventory available for release in the event of a fuel handling accident. Following sufficient decay occurring, the primary success path for mitigating the fuel handling accident no longer includes the functioning of the active containment systems. Therefore, the OPERABILITY requirements of the Technical Specifications are modified to reflect that water level and decay times are the primary success path for mitigating a fuel handling accident (which meets Criterion 3).
>
> To support this change in requirements during the handling of irradiated fuel, the OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features are deleted. The accidents postulated to occur during core alterations, in addition to fuel handling accidents, are: inadvertent criticality (due to a control rod removal error or continuous control rod withdrawal error during refueling) and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident, the proposed TS requirements omitting CORE ALTERATIONS is justified.

Also, the TS only allow the handling of irradiated fuel in the reactor vessel when the water level in the reactor cavity is at the high water level. Therefore, the proposed changes only affect containment requirements during periods of relatively low shutdown risk during refueling outages. Therefore, the proposed changes do not significantly increase the shutdown risk.

The TSTF-51, Revision 2, proposed changes have been previously reviewed by the NRC staff and found acceptable for inclusion in NUREG-1433 submittals. The June 29, 2005, submittal converting MNGP CTS to ITS, was based on NUREG-1433, therefore these additional changes to the MNGP ITS are applicable.

CREF Initiation TS Revisions

The additional changes to the Control Room Emergency Filtration (CREF) System Instrumentation TS and the Control Room Ventilation (CRV) TS (Enclosure 3) do not have any significant impact on the revision proposed by the previous AST submittal (Reference 1).

The inclusion of CREF System instrumentation initiation signals, one from Reactor Building Ventilation Exhaust - High, and the other from Refueling Floor Radiation - High, that initiate the CREF System will satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). These Functions are required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) and movement of recently irradiated fuel assemblies in the secondary containment. This is because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that control room dose limits are not exceeded. Due to radioactive decay, this Function is only required to initiate CREF during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

In the Reference 1 submittal, NMC stated this instrumentation was not necessary because MNGP Custom TS 3.10.D required that the reactor be shutdown for a minimum of 24 hours prior to movement of fuel within the reactor. With the conversion of MNGP to ITS, this TS requirement has been deleted. The addition of this instrumentation to the CREF TS will satisfy the requirements of 10 CFR 50.36 by providing CREF initiation during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours). As discussed in Reference 5, NMC does not propose, or intend, to handle recently irradiated fuel and therefore a FHA analysis was not performed for this scenario.

The MNGP Full Scope AST submittal (Reference 1) revised TS 3.17.A, "Control Room Ventilation System," such that the CRV System was not required to be OPERABLE during movement of irradiated fuel. The specification change did not address system OPERABILITY during movement of recently irradiated fuel because MNGP Custom TS 3.10.D required that the reactor be shutdown for a minimum of 24 hours prior to movement of fuel within the reactor. With the conversion of MNGP to ITS, TS 3.10.D has been deleted. Therefore, ITS 3.7.5 is being revised to specify that the CRV system must be OPERABLE during movement of recently irradiated fuel. This revision to the CRV TS will satisfy the requirements of 10 CFR 50.36. As stated above, NMC does not propose, or intend, to handle recently irradiated fuel and therefore a FHA analysis was not performed for this scenario.

Although the wording and format of the CREF and CRV TS have been revised, these changes preserve the intent of the Custom TS changes submitted in Reference 1 related to CREF initiation instrumentation and CRV OPERABILITY. Therefore, the changes to the TS are still acceptable for inclusion in the proposed MNGP TS change request.

Regulatory Evaluation

Changes to the MNGP TS are being proposed by NMC to incorporate the remainder of the TSTF-51, Revision 2 changes. These changes were not provided with the original Full Scope AST submittal (Reference 1) because the MNGP TS, at that time, did not contain these shutdown electrical power TS requirements.

The TSTF-51, Revision 2, proposed changes have been previously reviewed by the NRC staff and found acceptable for inclusion in NUREG-1433 submittals. The June 29, 2005, submittal converting MNGP CTS to ITS was based on NUREG-1433; therefore, these additional changes to the MNGP ITS are applicable.

The wording and format of the CREF and CRV TS have been revised; however, these changes preserve the intent of the Custom TS changes submitted in Reference 1 related to CREF initiation instrumentation and CRV System. Therefore, the additional changes to the CREF System Instrumentation TS and the CRV TS do not have a significant impact on the TS revisions proposed by the previous AST submittal (Reference 1).

NMC has reviewed the ITS conversion specifications in Enclosure 2 and the proposed TS changes in Enclosure 3 with respect to the no significant hazards and environmental considerations determinations previously submitted with the MNGP Full Scope AST LAR (Reference 1). It is concluded that the finding of no significant hazards is not affected by these proposed changes

References

1. NMC letter to NRC, "License Amendment Request: Full Scope Application of an Alternative Source Term," (ADAMS Accession No. ML052640366) dated September 15, 2005.

2. NRC letter to NMC, "Monticello Nuclear Generating Plant (MNGP) - Issuance of Amendment for the Conversion to the Improved Technical Specifications with Beyond-Scope Issues," (ADAMS Accession No. ML061070577) dated June 5, 2006.

3. NRC letter to NMC, "Monticello Nuclear Generating Plant (MNGP) - Issuance of Amendment Re: Use of the Alternative Source Term for the Postulated Fuel Handling Accident (ADAMS Accession No. ML060600572) dated April 24, 2006.

4. NMC letter to NRC, "License Amendment Request: Selective Scope Application of an Alternative Source Term Methodology for Re-evaluation of the Fuel Handling Accident" (ADAMS Accession No. ML041450022) dated April 29, 2004.

5. NMC Letter to NRC, "Response to Request for Additional Information Related to Technical Specifications Change Request to Apply Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident," dated January 11, 2005 (TAC No. MC3299)," dated January 20, 2005.

ENCLOSURE 2

Marked-Up Proposed Conversion Changes

(12 pages follow)

1.1 Definitions

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

Federal Guidance Report (FGR)-11, "Limiting Values of Radionuclide Intake and Air Concentration Factors for Inhalation, Submersion and Ingestion," September 1988, and FGR-12, "External Exposure to Radionuclides in Air, Water and Soil," September 1993.

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977.~~

LEAKAGE

LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE into the drywell, such as that from pump seals or valve packing that is captured and conducted to a sump or collecting tank; or

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

1, 2, and 3

APPLICABILITY: MODES 1 ~~and~~ 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of sodium pentaborate in solution not within limits of Figure 3.1.7-1 and Table 3.1.7-1 Equation 2, but available volume of sodium pentaborate solution is within limits of Table 3.1.7-1 Equation 1.	A.1 Restore concentration of sodium pentaborate in solution to within limits.	7 days
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1 Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	12 hours

<u>AND</u>

D.2 Be in MODE 4.

36 hours

3.3 INSTRUMENTATION

3.3.7.2 Mechanical Vacuum Pump Isolation Instrumentation

LCO 3.3.7.2 Four channels of the Main Steam Line Tunnel Radiation - High Function for the mechanical vacuum pump isolation shall be OPERABLE.

APPLICABILITY: MODES 1 and 2 with the mechanical vacuum pump in service and any main steam line not isolated.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status.	12 hours
	OR	
	A.2 ---------------NOTE--------------- Not applicable if inoperable channel is the result of an inoperable mechanical vacuum pump breaker or isolation valve. -------------------------------------- Place channel in trip.	12 hours
B. Mechanical vacuum pump isolation capability not maintained.	B.1 Restore mechanical vacuum pump isolation capability.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1 Isolate the mechanical vacuum pump.	12 hours
	OR	
	C.2 Isolate main steam lines.	12 hours
	OR	
	C.3 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

NOTE

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains mechanical vacuum pump isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.2.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 6.9 R/hour.	24 months
SR 3.3.7.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST, including mechanical vacuum pump breaker and isolation valves actuation.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ ~~2.0~~ μCi/gm.

0.2

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
0.2 2.0 A. Reactor coolant specific activity > ~~2.0~~ μCi/gm and ≤ ~~4.0~~ μCi/gm DOSE EQUIVALENT I-131.	------------------NOTE------------------ LCO 3.0.4.c is applicable. --	
	A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met. OR 2.0 Reactor Coolant specific activity > ~~4.0~~ μCi/gm DOSE EQUIVALENT I-131.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	B.2.1 Isolate all main steam lines.	12 hours
	OR	
	B.2.2.1 Be in MODE 3.	12 hours
	AND	
	B.2.2.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	------------------------------NOTE------------------------------ Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity is ≤ $\boxed{2.0}$ μCi/gm.	7 days

0.2

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

---NOTES---

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ------------NOTE------------ Only applicable to penetration flow paths with two PCIVs. -------------------------------- One or more penetration flow paths with one PCIV inoperable for reasons other than Condition D. or E	A.1 AND	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours except for main steam line AND 8 hours for main steam line

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ------------NOTE------------ Only applicable to penetration flow paths with two PCIVs. --------------------------------- One or more penetration flow paths with two PCIVs inoperable for reasons other than Condition D. or E	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. ------------NOTE------------ Only applicable to penetration flow paths with only one PCIV. --------------------------------- One or more penetration flow paths with one PCIV inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND	4 hours except for excess flow check valves (EFCVs) and penetrations with a closed system AND 72 hours for EFCVs and penetrations with a closed system

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more penetration flow paths with one or more 18 inch primary containment purge and vent valves not within purge and vent valve leakage limits.	D.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	24 hours
	AND		
	D.2	--------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ----------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment
F ~~E~~. Required Action and associated Completion Time of Condition A, B, C, ~~or~~ D , or E not met in MODE 1, 2, or 3.	~~E~~.1	Be in MODE 3.	12 hours
	AND		
	~~E~~.2	Be in MODE 4.	36 hours

F

E. One or more MSIVs with leakage rate not within limits.	E.1	Restore leakage rate to within limits.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. Required Action and associated Completion Time of Condition A or B not met for PCIV(s) required to be OPERABLE during MODE 4 or 5. G	F.1	Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	OR		
	F.2	Initiate action to restore valve(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.1	-----------------------------NOTE----------------------------- Not required to be met when the 18 inch primary containment purge and vent valves are open for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open. -- Verify each 18 inch primary containment purge and vent valve is closed.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 9.9 seconds.	24 months
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV actuates on a simulated instrument line break to restrict flow to ≤ 2 gpm.	24 months
SR 3.6.1.3.9	Verify each 18 inch primary containment purge and vent valve is blocked to restrict the valve from opening > 40°.	24 months
SR 3.6.1.3.10	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.11	Perform leakage rate testing for each 18 inch primary containment purge and vent valve with resilient seals.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.12	Verify leakage rate through each MSIV is: (a) ≤ 100 scfh when tested at ≥ 42 psig (P_a); or (a) ≤ 77 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.13	Verify leakage rate through the main steam pathway is: (a) ≤ 200 scfh when tested at ≥ 42 psig (P_a); or (b) ≤ 154 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

5.5 Programs and Manuals

5.5.10 Safety Function Determination Program (SFDP) (continued)

3. A required system redundant to the support system(s) for the supported systems described in Specifications 5.5.10.b.1 and 5.5.10.b.2 above is also inoperable.

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

5.5.11 Primary Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exception: s

1. The Type A testing Frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "at least once per 10 years based on acceptable performance history" is modified to be "at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in March 1993. ;

2. The main steam line pathway leakage contribution is excluded from the sum of the leakage rates from Type B and C tests specified in Section III.B of 10 CFR 50, Appendix J, Option B, Section 6.4.4 of ANSI/ANS 56.8-1994, and Section 10.2 of NEI 94-01, Rev. 0; and

3. The main steam line pathway leakage contribution is excluded from the overall integrated leakage rate from Type A tests specified in Section III.A of 10 CFR 50, Appendix J, Option B, Section 3.2 of ANSI/ANS 56.8-1994, and Section 8.0 and 9.0 of NEI 94-01, Rev. 0.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42 psig. The containment design pressure is 56 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 1.2% of containment air weight per day.

d. Leakage rate acceptance criteria are:

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

ENCLOSURE 3

Marked-Up Other Proposed Changes

(9 pages follow)

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Filtration (CREF) System Instrumentation

The CREF System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

LCO 3.3.7.1 ~~One channel per trip system of the Control Room Air Inlet Radiation - High Function shall be OPERABLE.~~

According to Table 3.3.7.1-1.

APPLICABILITY: ~~MODES 1, 2, and 3,~~
~~During movement of recently irradiated fuel assemblies in the secondary containment,~~
~~During operations with a potential for draining the reactor vessel (OPDRVs).~~

ACTIONS

---NOTE---

Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.		
	A.1 Place the associated CREF subsystem in the pressurization mode of operation. OR A.2 Declare associated CREF subsystem inoperable.	1 hour 1 hour

B.

	A.1 Declare associated CREF subsystem inoperable. AND A.2 Place channel in trip.	1 hour from discovery of loss of CREF initiation capability in both trip systems 12 hours
B. Required Action and associated Completion Time not met.		

1. Refer to Table 3.3.7.1-1 to determine which SRs apply for each CREF System Function.

SURVEILLANCE REQUIREMENTS

S

-----NOTE-----

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to

6 ~~8~~ hours provided the associated Function maintains CREF System initiation capability.

SURVEILLANCE	FREQUENCY
SR 3.3.7.1.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.7.1.2 Perform CHANNEL FUNCTIONAL TEST.	92 days
NEW SR — SR 3.3.7.1.3 Calibrate the trip unit.	92 days
NEW SR — SR 3.3.7.1.4 Perform CHANNEL CALIBRATION.	92 days
SR 3.3.7.1.~~3~~ 5 Perform CHANNEL CALIBRATION. ~~The Allowable Value shall be ≤ 2 mR/hour.~~	24 months
NEW SR — SR 3.3.7.1.6 Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

NEW TS TABLE

Table 3.3.7.1-1 (Page 1 of 1)
Control Room Emergency Filtration System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED ALLOWABLE CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	VALUE
1. Reactor Vessel Water Level - Low Low	1, 2, 3, (a)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ - 48 inches
2. Drywell Pressure - High	1, 2, 3	2	SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.6	≤ 2 psig
3. Reactor Building Ventilation Exhaust Radiation - High	1, 2, 3, (a), (b)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.6	≤ 100 mR/hr
4. Refueling Floor Radiation - High	1, 2, 3, (a), (b)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.6	≤ 100 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During movement of recently irradiated fuel assemblies in the secondary containment.

3.7 PLANT SYSTEMS

3.7.5 Control Room Ventilation System

LCO 3.7.5 Two control room ventilation subsystems shall be OPERABLE.

recently

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room ventilation subsystem inoperable.	A.1 Restore control room ventilation subsystem to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	-------------------NOTE------------------- LCO 3.0.3 is not applicable. --- C.1 Place OPERABLE control room ventilation subsystem in operation. OR	 Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
recently	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment. AND C.2.2 Initiate action to suspend OPDRVs.	Immediately Immediately
D. Two control room ventilation subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately
recently E. Two control room ventilation subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment or during OPDRVs.	--------------------NOTE-------------------- LCO 3.0.3 is not applicable. -- E.1 Suspend movement of irradiated fuel assemblies in the secondary containment. recently AND E.2 Initiate actions to suspend OPDRVs.	 Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each control room ventilation subsystem has the capability to remove the assumed heat load.	24 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems - Shutdown;" and

b. One emergency diesel generator (EDG) capable of supplying one division of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8.

recently

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	-------------------NOTE------------------ Enter applicable Condition and Required Actions of LCO 3.8.8, with one required division de-energized as a result of Condition A. ---	
	A.1 Declare affected required feature(s), with no offsite power available, inoperable.	Immediately
	<u>OR</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
recently	A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	AND	
	A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. One required EDG inoperable.	B.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	B.2 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
recently	AND	
	B.3 Initiate action to suspend OPDRVs.	Immediately
	AND	
	B.4 Initiate action to restore required EDG to OPERABLE status.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 — Division 1 or Division 2 125 VDC electrical power subsystem shall be OPERABLE to support one division of the DC Electrical Power Distribution System required by LCO 3.8.8, "Distribution Systems - Shutdown."

recently

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

--NOTE--
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required DC electrical power subsystem inoperable. recently	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend movement of irradiated fuel assemblies in the secondary containment. AND A.3 Initiate action to suspend operations with a potential for draining the reactor vessel. AND A.4 Initiate action to restore required DC electrical power subsystem to OPERABLE status.	Immediately Immediately Immediately Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems - Shutdown

LCO 3.8.8 The necessary portions of the AC and DC electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

recently

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC or DC electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
recently	A.2.2 Suspend handling of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	

ENCLOSURE 4

Marked-Up ITS Bases Changes

(54 pages follow)

BASES

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

4

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria,"~~ limits (Ref. ~~2~~). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

50.67, "Accident source term,"

REFERENCES

1. USAR, Section 1.2.2.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (revision specified in Specification 5.6.3).
3. NEDE-31152P, "General Electric Fuel Bundle Designs," Revision 8, April 2001.
4. 10 CFR ~~100~~. 50.67

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to USAR Section 4.2.1 (Ref. 1), the reactor vessel design pressure of 1250 psig was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation, with additional allowances to accommodate transients above the operating pressure without causing operation of the safety/relief valves. In addition, the reactor vessel was also designed for the transients that could occur during the design life.

During normal operation and anticipated operational occurrences (AOOs), RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2) for the pressure vessel. To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the reactor coolant pressure boundary (RCPB), reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 50.67, "Accident source term" ~~100, "Reactor Site Criteria"~~ (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure - High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1965 Edition, including Addenda through the summer of 1966 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig.

BASES

APPLICABLE SAFETY ANALYSES (continued)

A pressure of 1335 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to the USAS Nuclear Power Piping Code, Section B31.1, 1977 Edition, including Addenda through summer of 1978 (Ref. 6), for the piping, which permits a maximum pressure transient of 120% of design pressures of 1110 psig for piping communicating with the vessel steam space and 1136 psig for piping communicating with the bottom of the vessel. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 120% of design pressures of 1110 psig for piping communicating with the vessel steam space and 1136 psig for piping communicating with the bottom of the vessel. The most limiting of these allowances is the 120% of the piping communicating with the vessel steam space design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1332 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100,~~ 50.67, "Accident source term," ~~"Reactor Site Criteria,"~~ limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. USAR, Section 4.2.1.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. 10 CFR ~~100~~ 50.67
5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda summer of 1966

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (revision specified in Specification 5.6.3).

2. Letter from T.A. Pickens (BWROG) to G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.

3. USAR, Section 14.7.1.

4. NUREG-0979, Section 4.2.1.3.2, April 1983.

5. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

7. NEDO-10527, "Rod Drop Accident Analysis for Large BWRs," (including Supplements 1 and 2), March 1972.

8. ASME, Boiler and Pressure Vessel Code.

9. 10 CFR 100.11. 50.67

10. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram (ATWS).

INSERT A

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator determines the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron that produces a concentration of 660 ppm of natural boron in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping and with B-10 enrichment of ≥ 55.0 atom percent. This quantity of borated solution is the amount that is above the pump suction nozzle and accounts for wide range instrument accuracy. No credit is taken for the portion of the tank volume that cannot be injected.

3

INSERT B

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

a

3 and

BASES

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

INSERT B1

Furthermore, in

~~In~~ MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

A.1

If the concentration of sodium pentaborate in solution is not within limits of Figure 3.1.7-1 and Table 3.1.7-1 Equation 2 (ATWS design basis) but available volume of sodium pentaborate solution is within limits of Table 3.1.7-1 Equation 1 (original design basis), the concentration must be restored to within limits in 7 days. It is not necessary under these conditions to enter Condition C for both SLC subsystems inoperable since they are capable of performing their original design basis function. Because of the low probability of an event and the fact that the SLC System capability still exists for vessel injection under these conditions, the allowed Completion Time of 7 days is acceptable and provides adequate time to restore concentration to within limits.

, as well as providing suppression pool pH control following a LOCA

B.1

If one SLC subsystem is inoperable for reasons other than Condition A, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the ATWS design basis function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive (CRD) System to shut down the plant.

BASES

ACTIONS (continued)

C.1

If both SLC subsystems are inoperable for reasons other than Condition A, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

D.1

and MODE 4 within 36 hours

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

s

are

the required plant conditions

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1 and SR 3.1.7.2

SR 3.1.7.1 and SR 3.1.7.2 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 5°F margin will be maintained above the saturation temperature. The volume of sodium pentaborate solution requirements in Figure 3.1.7-1 and Table 3.1.7-1 Equation 1 will ensure both the original design basis and the ATWS design basis are met. Figure 3.1.7-1 can only be used if the B-10 enrichment in the storage tank is ≥ 55.0 atom percent. If the volume requirement of Table 3.1.7-1 Equation 1 is utilized for verification of volume requirements the concentration requirements for the original design basis can also be considered to be met. However, to verify the ATWS design basis requirements are met, Table 3.1.7-1 Equation 2 must be used to verify the concentration of sodium pentaborate solution requirements are met. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of volume and temperature.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests (laboratory analyses) on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used.

REFERENCES

1. 10 CFR 50.62.

3 ~~2~~. USAR, Section 6.6.1.1.

4. 10 CFR 50.67.
5. USAR Section 14.7.2.4.

2. NUREG-1465, "Accident Source Term for Light-Water Nuclear Power Plants, Final Report," February 1995.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. Each instrument volume is connected to a drain line with two valves in series for a total of four drain valves. Each header is connected to a vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR $\boxed{1\cancel{0}0}$ (Ref. 2) and

50.67

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR $\boxed{1\cancel{0}0}$ (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

50.67

SDV vent and drain valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of 30 seconds after receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis (Ref. 3). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3, "Control Rod OPERABILITY," overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 3.5.3.3.3.5.
2. 10 CFR 100 50.67
3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

in LCO 3.3.5.1, "Emergency Core Cooling Systems (ECCS) Instrumentation," and are not included in this LCO.

In general, the individual Functions are required to be OPERABLE in MODES 1, 2, and 3 consistent with the Applicability for LCO 3.6.1.1, "Primary Containment." Functions that have different Applicabilities are discussed below in the individual Functions discussion.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Main Steam Line Isolation

1.a. Reactor Vessel Water Level - Low Low

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level - Low Low Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level - Low Low Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Low Low supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level - Low Low Allowable Value is chosen to be the same as the ECCS Reactor Vessel Water Level - Low Low Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR $\boxed{100}$ limits.

50.67

This Function isolates the Group 1 valves.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four pressure switches that are connected to the MSL header close to the turbine stop valves. The pressure switches are arranged such that, even though physically separated from each other, each pressure switch is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 3).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow - High

Main Steam Line Flow - High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow - High Function is one of the Functions assumed in the analysis of the main steam line break (MSLB) (Ref. 2). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR $\boxed{100}$ limits.

50.67

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Primary Containment Isolation

2.a. Reactor Vessel Water Level - Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on low RPV water level supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ are not exceeded. 50.67 The Reactor Vessel Water Level - Low Function associated with isolation is implicitly assumed in the USAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level - Low signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Low Level - Low Allowable Value was chosen to be the same as the RPS Reactor Vessel Water Level - Low Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 2 drywell and sump isolation valves.

2.b. Drywell Pressure - High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ are not exceeded. The Drywell 50.67 Pressure - High Function, associated with isolation of the primary containment, is implicitly assumed in the USAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell Pressure - High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure - High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The RWCU Room Temperature - High Allowable Value is set low enough to detect a leak equivalent to 210 gpm.

This Function isolates the Group 3 valves.

5.c. Drywell Pressure - High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure - High Function, associated with isolation of the primary containment, is implicitly assumed in the USAR accident analysis as these leakage paths are assumed to be isolated post LOCA. 50.67

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell Pressure - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure - High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates the Group 3 valves.

5.d. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). SLC System initiation signals are initiated from the SLC initiation switch.

Two channels of the SLC System Initiation Function are available and are required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7, "Standby Liquid Control (SLC) System").

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

This Function isolates the Group 3 valves.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

administrative controls ensure that this flow path remains isolated to prevent unexpected loss of inventory via this flow path.

This Function isolates the Group 2 RHR shutdown cooling supply isolation valves.

Traversing Incore Probe System Isolation

7.a. Reactor Vessel Water Level - Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on low RPV water level supports actions to ensure that offsite dose limits of 10 CFR $\boxed{1\cancel{0}0}$ are not exceeded. 50.67
The Reactor Vessel Water Level - Low Function associated with isolation is implicitly assumed in the USAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level - Low signals are initiated from differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Two channels of Reactor Vessel Water Level - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can initiate an inadvertent isolation actuation. The isolation function is ensured by the manual shear valve in each penetration.

The Reactor Vessel Water Level - Low Allowable Value was chosen to be the same as the RPS Reactor Vessel Water Level - Low Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 2 TIP inboard isolation ball valves.

7.b. Drywell Pressure - High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR $\boxed{1\cancel{0}0}$ are not exceeded. The Drywell 50.67
Pressure - High Function, associated with isolation of the primary containment, is implicitly assumed in the USAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Filtration (CREF) System Instrumentation

BASES

BACKGROUND

Reactor Vessel Water Level - Low Low, Drywell Pressure - High, Reactor Building Ventilation Exhaust Radiation - High, or Refueling Floor Radiation - High

For both the Reactor Vessel Water Level - Low Low and Drywell Pressure - High Functions, the CREF System initiation logic receives input from four channels. The outputs from two channels for both Functions provide input into two trip systems. One channel must trip to trip a trip system and both trip systems must trip to initiate the CREF System (i.e., one-out-of-two taken twice logic arrangement). For both Reactor Building Ventilation Exhaust Radiation - High, and the Refueling Floor Radiation - High Functions, the CREF System initiation logic receives input from two channels. The outputs from each of the two channels for both Functions provide input into two trip systems. The logic for each Function in each trip system is arranged such that any channel can trip the trip system and initiate the CREF System.

The CREF System is designed to provide a radiologically controlled environment to ensure the habitability of the control room boundary (the main control room and portions of the first and second floors of the Emergency Filtration Train (EFT) building) for the safety of control room operators under all plant conditions. Two independent CREF subsystems are each capable of fulfilling the stated safety function. The instrumentation and controls for the CREF System automatically initiate action to isolate and pressurize the control room boundary to provide a radiologically controlled environment from which the unit can be safety operated following a Design Basis Accident (DBA).

In the event of a Control Room Air Inlet Radiation - High signal, the CREF System is automatically started in the pressurization mode. A system of dampers automatically isolates the control room boundary from untreated outside air. Outside air is taken in at the normal ventilation intake and is passed through one of the charcoal adsorber filter subsystems for removal of airborne radioactive particles. This air is then combined with return air from the control room boundary and passed through an exhaust/recirculation fan, which is then passed through the air handling unit into the control room boundary to maintain the control room boundary slightly pressurized with respect to the turbine building.

, either of which can initiate the CREF System

The CREF System instrumentation has two trip systems. One trip system isolates the control room boundary and initiates one CREF subsystem while the other trip system also isolates the control room boundary and initiates the other CREF subsystem. Each trip system receives input from one Control Room Air Inlet Radiation - High signal. The Control Room Air Inlet Radiation - High Function is arranged in a one-out-of-one logic. The channels include electronic equipment (e.g., relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CREF System initiation signal to the initiation logic.

trip units

APPLICABLE SAFETY ANALYSES

, LCO, and APPLICABILITY

The ability of the CREF System to maintain the habitability of the control room boundary is explicitly assumed for certain accidents as discussed in the USAR safety analysis (Ref. 1, 2, 3, and 4). CREF System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A.

the LOCA

the LOCA

CREF System instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

BASES

LCO

The OPERABILITY of the CREF System instrumentation is dependant upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.7.1-1. Each Function must have the required number of OPERABLE channels with their setpoints set within the specified Allowable Values, as shown in Table 3.3.7.1-1. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

~~The control room air inlet radiation monitors measure radiation levels exterior to the inlet ducting of the control room. A high radiation level may pose a threat to control room personnel; thus, automatically initiating the CREF System.~~

~~The Control Room Air Inlet Radiation - High Function consists of two independent monitors. Two channels of Control Room Air Inlet Radiation - High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude CREF System initiation. Each channel must have its setpoint within the specified Allowable Value of SR 3.3.7.1.3. The Allowable Value was selected to ensure protection of the control room personnel.~~

Allowable Values are specified for each CREF System Function specified in the Table.

~~The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.~~ Nominal trip setpoints are specified in ~~plant procedures.~~ the applicable setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

New paragraph

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. ~~The Allowable Value for the Control Room Air Inlet Radiation - High Function is set just above background to ensure that the control room operators are protected from increased radiation exposure.~~

INSERT CREF 1

APPLICABILITY

~~The Control Room Air Inlet Radiation - High Function is required to be OPERABLE in MODES 1, 2, and 3 and during OPDRVs and movement of recently irradiated fuel assemblies in the secondary containment, to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs), the probability of a LOCA is low; thus, the Function is not required. Also due to radioactive decay, this Function is only required to initiate the CREF System during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).~~

INSERT CREF 2

INSERT CREF 3

CREF 1 INSERT

The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. For Functions 1 and 2, the Allowable Values and nominal trip setpoints (NTSP) are derived, using the General Electric setpoint methodology guidance, as specified in the Monticello setpoint methodology. The Allowable Values are derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the NTSP allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events. For Functions 3 and 4, the Allowable Values and NTSP are based on engineering judgment.

CREF 2 INSERT

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions when the CREF System is required.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Reactor Vessel Water Level - Low Low

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The CREF System is initiated to maintain the habitability of the control room boundary. The Reactor Vessel Water Level - Low Low Function is one of the Functions assumed to be OPERABLE and capable of providing an initiation signal. The initiation of the CREF System on Reactor Vessel Water Level - Low Low supports actions to ensure that the habitability of the control room boundary is within the limits calculated in the safety analysis.

Reactor Vessel Water Level - Low Low signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level - Low Low Allowable Value was chosen to be the same as the High Pressure Coolant Injection/Reactor Core Isolation Cooling (HPCI/RCIC) Reactor Vessel Water Level - Low Low Allowable Value (LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," and LCO 3.3.5.2, "Reactor Core Isolation Cooling (RCIC) System Instrumentation"), since this could indicate that the capability to cool the fuel is being threatened.

The Reactor Vessel Water Level - Low Low Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is also required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) to ensure that control room dose limits are not exceeded if core damage occurs.

CREF 3 INSERT

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Drywell Pressure - High

High drywell pressure can indicate a break in the reactor coolant pressure boundary (RCPB). The CREF System is initiated to maintain the habitability of the control room boundary. The Drywell Pressure - High Function is one of the Functions assumed to be OPERABLE and capable of providing an initiation signal to ensure that control room doses are within the limits calculated in the safety analysis.

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell Pressure - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude performance of the isolation function.

The Allowable Value was chosen to be the same as the Reactor Protection System (RPS) Drywell Pressure - High Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") since this is indicative of a loss of coolant accident (LOCA).

The Drywell Pressure - High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This Function is not required in MODES 4 and 5 because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES.

3, 4. Reactor Building Ventilation Exhaust and Refueling Floor Radiation - High

High reactor building ventilation exhaust radiation or refuel floor radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When Reactor Building Ventilation Exhaust Radiation - High or Refueling Floor Radiation - High is detected the CREF System is initiated to limit the control room doses to less than the limits calculated in the safety analysis (Ref. 1).

The Reactor Building Ventilation Exhaust Radiation - High signals are initiated from radiation detectors that are located on the ventilation exhaust piping coming from the reactor building. The Refueling Floor Radiation - High signals are initiated for radiation detectors that are located to monitor the environment of the refuel floor area. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Two channels of Reactor Building Ventilation Exhaust Radiation - High Function and two channels of Refueling Floor Radiation - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Ventilation Exhaust Radiation - High and Refueling Floor Radiation - High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during OPDRVs and movement of recently irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel un-covering or dropped fuel assemblies) must be provided to ensure that control room dose limits are not exceeded. Due to radioactive decay, these Functions are only required to initiate the CREF System during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A Note has been provided to modify the ACTIONS related to CREF System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CREF System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CREF System instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide initiation signals and the redundancy of the CREF System design, an allowable out of service time of 12 hours has been shown to be acceptable to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function is still maintaining CREF System initiation capability. A Function is considered to be maintaining CREF System initiation capability when sufficient channels are OPERABLE or in trip such that one trip system will generate an initiation signal from the given Function on a valid signal. For Functions 1 and 2, this would require one trip system to have one channel per logic string OPERABLE or in trip (a logic string is the one-out-of-two portion of a one-out-of-two taken twice logic arrangement).

B.1 and B.2

~~A.1~~ and ~~A.2~~

With the Required Action and associated Completion Time not met

With ~~one or more channels inoperable~~, the associated CREF subsystem must be placed in the pressurization mode of operation (Required Action ~~A~~.1) to ensure that control room personnel will be protected in the event of a DBA. The method used to place the CREF subsystem in operation must provide for automatically reinitiating the subsystem upon restoration of power following a loss of power to the CREF subsystem(s). Alternately, if it is not desired to start the associated CREF subsystem, the CREF subsystem associated with the inoperable channels(s) must be declared inoperable within 1 hour (Required Action ~~A~~.2).

B

The 1 hour Completion Time is intended to allow the operator time to place the CREF subsystem in operation. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of channels, or for placing the associated CREF subsystem in operation.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each CREF System Instrumentation Function are located in the SRs column of Table 3.3.7.1-1.

The Surveillances are modified by a Note To indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to ~~8~~ 6 hours, provided the associated Function maintains CREF System initiation capability. Upon completion of the Surveillance, or expiration of the ~~8~~ 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. The Notes are based on the reliability analysis (Refs. 2 and 3) assumption of the average time required to perform channel surveillance. That analysis demonstrated the 6 hour testing allowance does not significantly reduce the probability that the CREF System will initiate when necessary.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.1

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 instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A 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 instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel status during normal operational use of the displays associated with channels required by the LCO.

SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the 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 shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of References 2 and 3.

SR 3.3.7.1.3

The calibration of trip units provides a check of the actual trip setpoints. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.7.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than the setting accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of References 2 and 3.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.$\boxed{3}$

4 and SR 3.3.7.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

~~The Frequency is based upon operating experience, which has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.~~

SR 3.3.7.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.7.4, "Control Room Emergency Filtration (CREF) System," overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. USAR, Section 14.7.2.4.3.
2. ~~USAR, Section 14.7.6.3.2~~
3. ~~USAR, Section 14.7.3.2.4~~
4. ~~USAR, Section 14.7.1.6~~

2. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
3. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

The Frequencies of SR 3.3.7.1.4 and SR 3.3.7.1.5 are based on the assumption of a 92 day and a 24 month Calibration interval, respectively, in the determination of the magnitude of equipment drift in the setpoint analysis.

B 3.3 INSTRUMENTATION

NEW TS BASES

B 3.3.7.2 Mechanical Vacuum Pump Isolation Instrumentation

BASES

BACKGROUND

The mechanical vacuum pump isolation instrumentation initiates a trip of the mechanical vacuum pump and isolation of the isolation valves following events in which main steam radiation monitors exceed a predetermined value. Tripping and isolating the mechanical vacuum pump limits control room and offsite doses in the event of a control rod drop accident (CRDA).

The mechanical vacuum pump isolation instrumentation includes sensors, relays and switches that are necessary to cause initiation of mechanical vacuum pump isolation. The channels include electronic equipment that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an isolation signal to the mechanical vacuum pump isolation logic.

The isolation logic consists of two independent trip systems, with two channels of the Main Steam Line Tunnel Radiation - High Function in each trip system. The outputs from two channels provide input into one trip system and the other two channels provide input into the other trip system. One channel must trip to trip a trip system and both trip systems must trip to initiate the mechanical vacuum pump isolation function (i.e., one-out-of-two taken twice logic arrangement). There are one mechanical vacuum pump breaker and two mechanical vacuum pump isolation valves associated with this Function.

APPLICABLE SAFETY ANALYSES

The mechanical vacuum pump isolation is assumed in the safety analysis for the CRDA (Ref. 1). The mechanical vacuum pump isolation instrumentation initiates an isolation of the mechanical vacuum pump to limit control room and offsite doses resulting from fuel cladding failure in a CRDA. An Allowable Value of 6.9 R/hr will ensure that 10 CFR 50, Appendix A, General Design Criterion (GDC) 19 limits (Ref. 2) will not be exceeded in the control room in the event of a CRDA.

The mechanical vacuum pump isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES NEW TS BASES

LCO

The OPERABILITY of the mechanical vacuum pump isolation instrumentation is dependent on the OPERABILITY of the individual Main Steam Line Tunnel Radiation - High Function instrumentation channels, which must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.7.2.3. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also includes the mechanical vacuum pump breaker and isolation valves. An Allowable Value is specified for the Main Steam Line Tunnel Radiation - High Function specified in the LCO. A nominal trip setpoint is specified in the setpoint calculations. The nominal trip setpoint is selected to ensure that the actual trip setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (i.e., main steam line tunnel radiation), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values and nominal trip setpoints (NTSP) are derived, using the General Electric setpoint methodology guidance, as specified in the Monticello setpoint methodology. The Allowable Values are derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the NTSP allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events.

BASES

NEW TS BASES

APPLICABILITY

The mechanical vacuum pump isolation is required to be OPERABLE in MODES 1 and 2 when the mechanical vacuum pump is in service (i.e., taking suction on the main condenser) and any main steam line not isolated, to mitigate the consequences of a postulated CRDA. In this condition, fission products released during a CRDA could be discharged directly to the environment. Therefore, the mechanical vacuum pump isolation is necessary to assure conformance with the radiological evaluation of the CRDA. In MODE 3, 4 or 5 the consequences of a control rod drop are insignificant, and are not expected to result in any fuel damage or fission product releases. In MODES 1 or 2 when the mechanical vacuum pump is not in operation or the main steam lines are isolated, fission product releases via this pathway would not occur.

ACTIONS

A Note has been provided to modify the ACTIONS related to mechanical vacuum pump isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable mechanical vacuum pump isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable mechanical vacuum pump isolation instrumentation channel.

A.1 and A.2

With one or more channels inoperable, but with mechanical vacuum pump isolation capability maintained (refer to Required Action B.1 Bases), the mechanical vacuum pump isolation instrumentation is capable of performing the intended function. However, the reliability and redundancy of the mechanical vacuum pump isolation instrumentation is reduced, such that a single failure in one of the remaining channels could result in the inability of the mechanical vacuum pump isolation instrumentation to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the low probability of extensive number of inoperabilities affecting multiple channels, and the low probability of an event requiring the initiation of the mechanical vacuum pump isolation, 12 hours has been shown to be acceptable (Ref. 3) to permit restoration of any inoperable channel to OPERABLE status (Required Action A.1). Alternately, the inoperable channel may be placed in trip (Required Action A.2), since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted,

BASES

NEW TS BASES

ACTIONS (continued)

placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable mechanical vacuum pump breaker or isolation valve, since this may not adequately compensate for the inoperable mechanical vacuum pump breaker or isolation valve (e.g., the breaker may be inoperable such that it will not trip). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in loss of condenser vacuum), or if the inoperable channel is the result of an inoperable mechanical vacuum pump breaker or isolation valve, Condition C must be entered and its Required Actions taken.

B.1

Condition B is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels result in the Function not maintaining mechanical vacuum pump isolation capability. The Function is considered to be maintaining mechanical vacuum pump isolation capability when sufficient channels are OPERABLE or in trip such that the mechanical vacuum pump isolation instruments will generate a trip signal from a valid Main Steam Line Tunnel Radiation - High signal, and the mechanical vacuum pump will trip or the isolation valves will close. This requires one channel of the Function in each trip system to be OPERABLE or in trip, and the mechanical vacuum pump breaker or isolation valves to be OPERABLE.

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

With any Required Action and associated Completion Time of Condition A or B not met, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours (Required Action C.3). Alternately, the mechanical vacuum pump may be isolated (Required Action C.1) since this action performs the intended function of the instrumentation. An additional option is provided to isolate the main steam lines (Required Action C.2), which may allow operation to continue. Isolating the main steam lines effectively provides an equivalent level of protection by precluding fission product transport to the condenser. This isolation is accomplished by isolation of all main steam lines and all main steam line drains that bypass the main steam isolation valves.

BASES

NEW TS BASES

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains mechanical vacuum pump isolation trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analyses (Ref. 3) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the mechanical vacuum pump will isolate when necessary.

SR 3.3.7.2.1

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 instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A 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 combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based on the CHANNEL CHECK Frequency requirement of other instrumentation. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the required channels of this LCO.

SR 3.3.7.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 1.

BASES NEW TS BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.2.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.7.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the mechanical vacuum pump breaker and actuation of the associated isolation valves are included as part of this Surveillance, to provide complete testing of the assumed safety function. Therefore, if the breaker is incapable of operating or an isolation valve is incapable of closing, the instrument channel would be inoperable.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. USAR Section 14.7.1
2. 10 CFR 50, Appendix A, GDC 19.
3. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR ~~100~~ (Ref. 1).

50.67

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR ~~100~~ limit. 50.67

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the USAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

Total Effective Dose Equivalent (TEDE)

This MSLB release forms the basis for determining offsite and control room doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour ~~thyroid and whole body~~ doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed 10% of the dose guidelines of

50.67

10 CFR ~~100~~. The limits on the specific activity of the primary coolant also

TEDE

ensure the ~~thyroid~~ dose to control room operators, resulting from a MSLB outside containment during steady operation, will not exceed the limits of 10 CFR 50, Appendix A, GDC 19 (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)

based on plant specific analysis of the radiological consequences of a MSLB accident (Ref. 2).

The limit on specific activity is ~~a value from a parametric evaluation of typical site locations. This limit is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.~~

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

LCO

The specific iodine activity is limited to ≤ ~~2.0~~ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR ~~100~~ 50.67 limits and is less than 10 CFR 50, Appendix A, GDC 19 (Ref. 3) limits.

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ ~~4.0~~ 2.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

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. This allowance 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, B.2.1, B.2.2.1, and B.2.2.2

0.2

2.0

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 2.0 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least once every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than ~~a small fraction of~~ the ~~requirements~~ of 10 CFR ~~100~~ and 10 CFR 50, Appendix A, GDC 19 (Ref. 3) during a postulated MSLB accident.

limits

50.67

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR ~~100.11~~ 50.67

2. USAR, Section 14.7.3.

3. 10 CFR 50, Appendix A, GDC 19.

BASES

BACKGROUND (continued)

high pressure from reaching the Standby Gas Treatment (SGT) System filter trains in the unlikely event of a loss of coolant accident (LOCA) during venting. When the 18 inch purge and vent valves are opened, they must be aligned to the reactor building plenum and vent. The 18 inch purge and vent valves are capable of closing in the environment of a LOCA.

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material for which the consequences are mitigated by PCIVs are a LOCA and a main steam line break (MSLB) (Refs. 2 and 3, respectively). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or close within the required isolation times following event initiation. This ensures that potential paths to the environment through PCIVs (including primary containment purge and vent valves) are minimized. The radiological consequences of the LOCA ~~is~~ are discussed in Reference 4 while the radiological consequences of the MSLB ~~is~~ are discussed in Reference 5. The ~~whole body~~ offsite dose consequences is most limiting ~~in the LOCA~~ while the thyroid offsite dose consequences is most limiting ~~in the MSLB~~ event. The control room dose consequences is ~~most limiting~~ in the LOCA event. The MSIVs are required to close within 3 to 9.9 seconds. The 3 second closure time is assumed in the MSIV closure (the most severe overpressurization transient) analysis (Ref. 6). The 9.9 second closure time is assumed in the MSLB analysis (The analysis assumes a total time of 10.5 seconds of which 0.6 seconds is assumed for instrument response). The safety analyses do not make any explicit assumptions concerning the purge and vent valves position at event initiation. However, the purge and vent valves have been designed to close prior to the onset of fuel failure following a LOCA. Likewise, it is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

The DBA analysis assumes that isolation of the primary containment is complete and leakage is terminated, except for the maximum allowable leakage rate, L_a, prior to fuel damage.

BASES

SURVEILLANCE REQUIREMENTS (continued)

administrative controls. Allowing verification by administrative controls is considered acceptable since the primary containment is inerted and access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these PCIVs, once they have been verified to be in their proper position, is low. A second Note has been included to clarify that PCIVs that are open under administrative controls are not required to meet the SR during the time that the PCIVs are open. These controls consist of stationing a dedicated individual at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

SR 3.6.1.3.4

The traversing incore probe (TIP) shear isolation valves are actuated by explosive charges. Surveillance of explosive charge continuity provides assurance that TIP valves will actuate when required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience that has demonstrated the reliability of the explosive charge continuity.

SR 3.6.1.3.5

Verifying the isolation time of each power operated, automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.6. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The Frequency of this SR is 24 months.

SR 3.6.1.3.6

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA and transient analyses. This ensures that the calculated radiological consequences of these events remain within
50.67 10 CFR 100 limits. The Frequency of this SR is 24 months.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.10

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 24 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.11

For the 18 inch primary containment purge and vent valves with resilient seals, leakage rate testing consistent with the test requirements of 10 CFR 50, Appendix J, Option B (Ref. 8), is required to ensure OPERABILITY. The Frequency of this SR is in accordance with the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.12

The Alternate Source Term analyses are based on leakage that is less than the specified leakage rate. *Leakage through each MSIV must be ≤ 100 scfh when tested at ≥ 42 psig (P_a) or* ≤ 77 scfh when tested at ≥ 25 psig (Pt). This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency of this SR is in accordance with the Primary Containment Leakage Rate Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

The Alternate Source Term analyses are based on leakage that is less than the specified leakage rate. Leakage through the main steam pathway (i.e., the four main steam lines and the main steam line drains) must be ≤ 200 scfh when tested at ≥ 42 psig (P_a) or ≤ 154 scfh when tested at ≥ 25 psig (Pt). This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

REFERENCES

1. USAR, Table 5.2-3a.
2. USAR, Section 14.7.2.
3. USAR, Section 14.7.3.
4. USAR, Section 14.7.2.4.
5. USAR, Section 14.7.3.2.
6. USAR, Section 14.5.1.
7. USAR, Table 5.2-3b.
8. 10 CFR 50, Appendix J, Option B.
9. Letter from L. O. Mayer (NSP) to J. F. O'Leary (NRC), dated July 27, 1973.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Emergency Filtration (CREF) System

BASES

BACKGROUND

The CREF System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of CREF System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of outside supply air. Each subsystem consists of a low efficiency filter, an electric heater, a high efficiency particulate air (HEPA) filter, two activated charcoal adsorber sections, a second HEPA filter, an emergency filter fan, an air handling unit (excluding the condensing unit), an exhaust/recirculation fan, and the associated ductwork and dampers. Low efficiency filters and HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

Reactor Vessel Water Level - Low Low, Drywell Pressure - High, Refueling Floor Radiation - High or Reactor Building Ventilation Exhaust Radition - High initiation signal

The CREF System is a standby system, parts of which also operate during normal unit operations to maintain the control room boundary environment. Upon receipt of a ~~Control Room Air Inlet Radiation - High initiation signal (indicative of conditions that could result in radiation exposure to control room personnel)~~, the CREF System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room boundary (the main control room and portions of the first and second floors of the Emergency Filtration Train (EFT) building). A system of dampers isolates the control room boundary from untreated outside air. Outside air is taken in at the normal ventilation intake and is passed through one of the charcoal adsorber filter subsystems for removal of airborne radioactive particles. This air is then combined with return air from the control room boundary and passed through an exhaust/recirculation fan, which is then passed through the air handling unit into the control room boundary.

TEDE

The CREF System is designed to maintain the control room boundary environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem ~~whole body dose or its equivalent to any part of the body~~. A single CREF subsystem will pressurize the control room boundary to prevent infiltration of air from surrounding buildings. CREF System operation in maintaining control room habitability is discussed in the USAR, Section~~s~~ 14.7.2.4.3~~, 14.7.6.3.2, 14.7.3.2.4, and 14.7.1.6~~ (Ref. 1~~, 2, 3, and 4, respectively~~).

BASES

APPLICABLE SAFETY ANALYSES

and is required during movement of recently irradiated fuel in the secondary containment

LOCA analysis

The ability of the CREF System to maintain the habitability of the control room boundary is an explicit assumption for the safety analyses presented in the USAR, Sections 14.7.2.4.3, 14.7.6.3.2, 14.7.3.2.4, and 14.7.1.6 (Ref. 1, 2, 3, and 4, respectively). The pressurization mode of the CREF System is assumed to operate following a loss of coolant accident, fuel handling accident involving handling recently irradiated fuel (i.e., fuel ~~that has~~ occupied ~~part of a~~ critical reactor core within the previous 24 hours), main steam ~~line break, and~~ control rod drop accident, as discussed in the USAR, Sections 14.7.2.4.3, 14.7.6.3.2, 14.7.3.2.4, and 14.7.1.6 (Ref. 1, 2, 3, and 4, respectively). No single active failure will prevent the CREF System from performing its safety function.

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

LCO

Two redundant subsystems of the CREF System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA.

LOCA

The CREF System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Emergency filter fan, exhaust/recirculation fan, and air handling unit (excluding the condenser unit) are OPERABLE;

b. Low efficiency filter, HEPA filters, and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions; and

c. Heater, ductwork, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the main control room boundary to be opened intermittently under administrative controls. 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 consist of stationing a dedicated individual at the opening who is in continuous communication with the main control room. This individual will have a method to rapidly close the opening when a need for main control room boundary isolation is indicated.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the CREF System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release. LOCA

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREF System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with potential for draining the reactor vessel (OPDRVs); and

b. During movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the CREF System is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A.1

With one CREF subsystem inoperable, the inoperable CREF subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CREF subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced CREF System capability. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1

If the main control room boundary is inoperable in MODE 1, 2, or 3, the CREF subsystems cannot perform their intended functions. Actions must be taken to restore an OPERABLE main control room boundary within 24 hours. During the period that the main control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of 10 CFR 50, Appendix A, GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the main control room boundary.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.2

This SR verifies that the required CREF testing is performed in accordance with the Ventilation Filter Testing Program (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.

SR 3.7.4.3

This SR verifies that on an actual or simulated initiation signal, each CREF subsystem starts and operates. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.7.1, "Control Room Emergency Filtration (CREF) Instrumentation," overlaps this SR to provide complete testing of the safety function. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.4.4

This SR verifies the integrity of the control room boundary and the assumed inleakage rates of potentially contaminated air. The control room boundary positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the CREF System. During the emergency mode of operation, the CREF System is designed to slightly pressurize the control room boundary with respect to adjacent areas to prevent unfiltered inleakage. The CREF System is designed to maintain this positive pressure at a flow rate of ≤ 1100 cfm to the control room boundary in the pressurization mode. The Frequency of 24 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration systems SRs.

REFERENCES

1. USAR, Section 14.7.2.4.3.
2. USAR, ~~Section~~ 14.7.6.3.2.
3. USAR, ~~Section~~ 14.7.3.2.4.
4. USAR, ~~Section~~ 14.7.1.6.

BASES

LCO

Two independent and redundant subsystems of the Control Room Ventilation System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Control Room Ventilation System is considered OPERABLE when the individual components necessary to maintain the control room boundary temperature are OPERABLE in both subsystems. These components include the cooling coils, fans, compressors, ductwork, dampers, and associated instrumentation and controls.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room Ventilation System must be OPERABLE to ensure that the control room boundary temperature will not exceed equipment OPERABILITY limits following control room isolation.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room Ventilation System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with a potential for draining the reactor vessel (OPDRVs); and

recently

b. During movement of irradiated fuel assemblies in the secondary containment.

INSERT C

ACTIONS

A.1

With one control room ventilation subsystem inoperable, the inoperable control room ventilation subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room ventilation subsystem is adequate to perform the control room boundary air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room boundary air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate safety and nonsafety cooling methods.

BASES

ACTIONS (continued)

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable control room ventilation subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

, although not feasible,

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

recently

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

recently

recently

During movement of irradiated fuel assemblies in the secondary containment or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE control room ventilation subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

recently

If applicable, movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

BASES

ACTIONS (continued)

D.1

If both control room ventilation subsystems are inoperable in MODE 1, 2, or 3, the Control Room Ventilation System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately.

, although not feasible,

E.1 and E.2

recently

The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

recently

recently

During movement of irradiated fuel assemblies in the secondary containment or during OPDRVs, with two control room ventilation subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

recently

If applicable, handling of irradiated fuel in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room boundary heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 24 month Frequency is appropriate since significant degradation of the Control Room Ventilation System is not expected over this time period.

REFERENCES

1. USAR, Section 6.7.

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensible gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser. The radioactivity of the remaining gaseous mixture is monitored at the outlet of the offgas condenser prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of an event that inadvertently releases the main condenser effluent directly to the environment without treatment. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR $\boxed{\cancel{100}}$ (Ref. 1).

The main condenser offgas limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). 50.67

LCO

To ensure compliance with the assumptions of an event that inadvertently releases the main condenser effluent directly to the environment without treatment, the fission product release rate must be ≤ 260 mCi/second after decay of 30 minutes.

APPLICABILITY

The LCO is applicable when steam is being exhausted to the main condenser and the resulting noncondensibles are being processed via the Main Condenser Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, steam is not being exhausted to the main condenser and the requirements are not applicable.

ACTIONS

A.1

If the offgas radioactivity rate limit is exceeded, 72 hours is allowed to restore the gross gamma activity rate to within the limit. The 72 hour Completion Time is reasonable, based on engineering judgment, the time required to complete the Required Action, the large margins associated

BASES

ACTIONS (continued)

with permissible dose and exposure limits, and the low probability of an event that inadvertently releases the main condenser effluent directly to the environment without treatment.

B.1, B.2, B.3.1, and B.3.2

If the gross gamma activity rate is not restored to within the limits in the associated Completion Time, all main steam lines or the SJAE must be isolated. This isolates the Main Condenser Offgas System from the source of the radioactive steam. The main steam lines are considered isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain valve in each drain line is closed. The 12 hour Completion Time is reasonable, based on operating experience, to perform the actions from full power conditions in an orderly manner and without challenging unit systems.

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable, based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.

REFERENCES

1. 10 CFR ~~100~~. 50.67

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment ensures that: recently

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

INSERT D

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving handling irradiated fuel.

recently

In general, when the unit is shut down the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or loss of all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and corresponding stresses result in the probabilities of occurrences significantly reduced or eliminated, and minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, and 3, various deviations from the analysis assumptions and design requirements are allowed within the ACTIONS. This allowance is in recognition that certain testing and maintenance activities must be conducted, provided an acceptable level of risk is not exceeded. During MODES 4 and 5, performance of a significant number of required testing and maintenance activities is also required. In MODES 4 and 5, the activities are generally planned and administratively controlled. Relaxations from typical MODES 1, 2, and 3 LCO requirements are acceptable during shutdown MODES, based on:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operation MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODES 1, 2, and 3 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability of supporting systems necessary for avoiding immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite (emergency diesel generator (EDG)) power.

AC Sources - Shutdown satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.8, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE EDG, associated with a Distribution System 4.16 kV essential bus required OPERABLE by LCO 3.8.8, ensures that a diverse power source is available for providing electrical power support assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and EDG ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling irradiated fuel and inadvertent reactor vessel draindown). Automatic initiation of the required EDG during shutdown conditions is specified in LCO 3.3.5.1, "ECCS Instrumentation," and LCO 3.3.8.1, "LOP Instrumentation."

recently

The qualified offsite circuit(s) must be capable of maintaining rated frequency and voltage while connected to their respective 4.16 kV essential bus, and of accepting required loads during an accident. The primary AC electrical power distribution subsystem for each division consists of a 4.16 kV essential bus (essential bus 15 for Division 1 and essential bus 16 for Division 2) having several offsite sources of power available. One offsite circuit consists of incoming disconnects to the

BASES

LCO (continued)

2R transformer, associated 2R transformer, and the respective circuit path including buses and feeder breakers to all 4.16 kV essential buses required by LCO 3.8.8. The second circuit consists of incoming disconnects to the 1R transformer, associated 1R transformer, and the respective circuit path including buses and feeder breakers to all 4.16 kV essential buses required by LCO 3.8.8. The third qualified offsite circuit consists of incoming disconnects to the 1AR transformer, associated 1AR transformer, and the respective circuit path including feeder breakers to all 4.16 kV essential buses required by LCO 3.8.8.

The required EDG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective 4.16 kV essential bus on detection of bus undervoltage, and accepting required loads. This sequence must be accomplished within 10 seconds. Each EDG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the 4.16 kV essential buses. These capabilities are required to be met from a variety of initial conditions such as EDG in standby with engine hot and EDG in standby with engine at ambient conditions. Additional EDG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the EDG to reject a load equivalent to its associated single largest post-accident load.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for EDG OPERABILITY. In addition, proper sequence operation is an integral part of offsite circuit OPERABILITY since its inoperability impacts the ability to start and maintain energized loads required OPERABLE by LCO 3.8.8.

The necessary portions of the Emergency Diesel Generator - Emergency Service Water System capable of providing cooling to the required EDG are also required.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment to provide assurance that:

recently

a. Systems providing adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident involving handling irradiated fuel;

recently

are available

BASES

APPLICABILITY (continued)

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

should be accounted for

AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

ACTIONS

although not feasible,

recently

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement [~~can occur~~] in MODE 1, 2, or 3, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, or 3 would require the unit to be shutdown unnecessarily.

recently

recently

, although not feasible,

A.1

An offsite circuit is considered inoperable if it is not available to one required division. If two or more 4.16 kV essential buses are required per LCO 3.8.8, one division with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, irradiated fuel movement, and operations with a potential for draining the reactor vessel. By the allowance of the option to declare required features inoperable with no offsite power available, appropriate restrictions can be implemented in accordance with the affected required feature(s) LCOs' ACTIONS.

recently

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required divisions, the option still exists to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required EDG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in the secondary containment, and activities that could result in inadvertent draining of the reactor vessel.

recently

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in USAR, Chapter 14 (Ref. 1), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the emergency diesel generators (EDGs), emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 and during movement of irradiated fuel assemblies ensures that: recently

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving handling irradiated fuel.

recently

INSERT F

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The DC Sources - Shutdown satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The Division 1 or Division 2 125 VDC electrical power subsystem consisting of one 125 V battery, one battery charger, and the corresponding control equipment and interconnecting cabling is required to be OPERABLE to support one division of the DC electrical power distribution subsystem(s) required OPERABLE by LCO 3.8.8, "Distribution Systems - Shutdown." This requirement ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling
recently
irradiated fuel and inadvertent reactor vessel draindown).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment provide assurance that:
recently

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Required features needed to mitigate a fuel handling accident involving handling irradiated fuel are available;
recently

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

should be accounted for

ACTIONS

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement [~~can occur~~] in MODE 1, 2, or 3, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, or 3 would require the unit to be shutdown unnecessarily.

recently

although not feasible,

recently

although not feasible,

recently

A.1, A.2, A.3, and A.4

If the required Division 1 or Division 2 125 VDC electrical power subsystem is inoperable, the minimum required DC power sources are not available. Therefore, suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies in the secondary containment, and any activities that could result in inadvertent draining of the reactor vessel is required.

recently

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystem and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystem should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.3. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the AC and DC electrical power distribution system is provided in the Bases for LCO 3.8.7, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in USAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC and DC electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Emergency Core Cooling Systems and Reactor Core Isolation Cooling System, and containment design limits are not exceeded.

The OPERABILITY of the AC and DC electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5,
recently
and during movement of irradiated fuel assemblies in the secondary containment ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving handling irradiated fuel. INSERT G
recently

The Distribution Systems - Shutdown satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of Technical Specifications required systems, equipment, and components - both specifically addressed by their own LCO, and implicitly required by the definition of OPERABILITY.

BASES

LCO (continued)

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling irradiated fuel and inadvertent reactor vessel draindown).

recently

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment provide assurance that:

recently

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident involving handling irradiated fuel;

recently

are available

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

should be accounted for

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

ACTIONS

although not feasible,

recently

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement [~~can occur~~] in MODE 1, 2, or 3, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, or 3 would require the unit to be shutdown unnecessarily.

recently

recently

although not feasible,

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant divisions of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem division may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, irradiated fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required

recently

BASES

ACTIONS (continued)

recently

features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made, (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in the secondary containment, and any activities that could result in inadvertent draining of the reactor vessel).

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal-shutdown cooling (RHR-SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR-SDC ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring RHR-SDC inoperable, which results in taking the appropriate RHR-SDC ACTIONS.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the AC and DC electrical power distribution subsystems are functioning properly, with the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the electrical power distribution subsystems, as well as other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. USAR, Chapter 14.