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RS-01-022

February 15, 2001

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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Revision D to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications

Reference:

1) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated March 3, 2000

2) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision A to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated June 5, 2000

3) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision B to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated September 1, 2000

A001

4) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision C to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated December 18, 2000

5) Letter from R. M. Krich (Exelon Generation Company) to U.S. NRC, "Draft Safety Evaluations for the Conversion to Improved Standard Technical Specifications," dated February 2, 2001

In Reference 1, in accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," we proposed to amend Appendix A, Technical Specifications (TS) of Facility Operating License Nos. DPR-19, DPR-25, NPF-11, NPF-18, DPR-29 and DPR-30 for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The proposed changes revise the Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, current Technical Specifications (CTS) to a format and content consistent with NUREG-1433, Revision 1, "Standard Technical Specifications for General Electric Plants, BWR 4," and NUREG-1434, Revision 1, "Standard Technical Specifications for General Electric Plants, BWR 6," as applicable. References 2, 3 and 4 subsequently supplemented the proposed amendment.

This letter submits further modifications for the proposed changes as Attachments 1, 2, and 3 to this letter. Each modification and its source is described in the "Summary of Changes" section of Attachments 1, 2, and 3. The sources include the changes made based upon discussions in meetings and telephone conferences between Exelon Generation Company (EGC), LLC, formerly Commonwealth Edison Company, personnel and the staff, and minor corrections to various sections of the proposed change. In addition, we provided comments concerning the draft Safety Evaluations (SEs) for the conversion to Improved Standard Technical Specifications in Reference 5. Attachment 4 provides a new comment on the draft SEs that was not previously submitted.

The modifications have been reviewed and approved by the respective Plant Operations Review Committees and the Nuclear Safety Review Board in accordance with the Quality Assurance Program.

EGC is notifying the State of Illinois of these modifications to previously submitted license amendment requests by transmitting a copy of this letter, including attachments and enclosures, to the designated state official.

Should you have any questions concerning this submittal, please contact Mr. J. V. Sipek at (630) 663-3741.

Respectfully,

R. M. Krich
Director-Licensing
Mid-West Regional Operating Group

Attachments: Affidavit
Attachment 1 - Revision D to Dresden Improved Technical Specifications Document
Attachment 2 - Revision D to LaSalle Improved Technical Specifications Document
Attachment 3 - Revision D to Quad Cities Improved Technical Specifications Document
Attachment 4 – Additional Comment on Draft Safety Evaluation

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Dresden Nuclear Power Station
w/o Attachments 2 and 3
NRC Senior Resident Inspector - LaSalle County Station
w/o Attachments 1 and 3
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
w/o Attachments 1 and 2
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON (COMED) COMPANY	)	**Docket Nos.**
DRESDEN NUCLEAR POWER STATION - UNITS 2 and 3	)	**50- 237and 50-249**
LASALLE COUNTY STATION - UNITS 1 and 2	)	**50- 373 and 50-374**
QUAD CITIES NUCLEAR POWER STATION - UNITS 1 and 2	)	**50- 254 and 50-265**

SUBJECT: **Revision D to Request for Amendment to Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Director, Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 15th day of February, 2001.

"OFFICIAL SEAL"
V. LYNN BERNHAGEN
Notary Public, State of Illinois
My Commission Exp. 05/30/2004

V. Lynn Bernhagen
Notary Public

ATTACHMENT 1

Revision D to Dresden Nuclear Power Station, Units 2 and 3 Proposed Improved Technical Specifications Submittal dated March 3, 2000

Revision D to Dresden Nuclear Power Station
Improved Technical Specifications Summary of Changes

This attachment provides a brief summary of the changes in Revision D of the proposed Improved Technical Specifications (ITS) submittal for Dresden Nuclear Power Station, Units 2 and 3. The original Technical Specifications amendment request (i.e., Revision 0) was submitted to the NRC by letter dated March 3, 2000, as revised in Revisions A, B, and C, submitted to the NRC by letters dated June 5, 2000, September 1, 2000, and December 18, 2000, respectively.

Changes committed to based on discussions with the NRC reviewers, minor technical changes, and editorial corrections are included in this revision.

Chapter 1.0

1. Typographical errors have been corrected (changed the word "page" to "pages" in the Dose Equivalent I-131 definition and changed the word "was" to "were" in the ISTS markup for Section 1.4). These changes affect ITS 1.1 page 1.1-3 and the Improved Standard Technical Specifications (ISTS) markup page 1.1-3 and 1.4-5.

Section 3.1

1. The word "boron" in the second Frequency in SR 3.1.7.5 has been changed to "sodium pentaborate" to be consistent with the words in the actual SR. This change affects the ISTS 3.1.7 markup page 3.1-21.

Section 3.2

1. The Bases for ITS 3.2.4 have been modified to include the information relocated by ITS Chapter 1.0 DOC LA.1. This change affects ITS 3.2.4 Bases page B 3.2.4-1 and the ISTS Bases markup insert page B 3.2-14.

Section 3.3

1. Typographical/editorial corrections have been made to ITS 3.3.1.1 (The addition of the letter "s" to the word Function" in ACTIONS Note 2, the identification of the correct Function number in SR 3.3.1.1.17 Note 2, and the deletion of the second "the" in SR 3.3.1.1.19 Note 2). These changes affect ITS 3.3.1.1 pages 3.3.1.1-1 and 3.3.1.1-7 and the ISTS markup pages 3.3-1, 3.3-5, and 3.3-6.

2. The Allowable Values for ITS 3.3.1.1 Functions 2.b (clamped Allowable Value only) and 2.c have been changed to be consistent with the Allowable Value for the same Function in the LaSalle ITS. In addition, the Allowable Values for ITS 3.3.1.1 Function 3 and SR 3.3.5.2.2 have been changed to prevent inadvertent scrams/initiations. The new Allowable Values are consistent with the current setpoint calculations. The change affects ITS 3.3.1.1 pages 3.3.1.1-8 and 3.3.1.1-9, ITS 3.3.5.2 page 3.3.5.2-2, and the ISTS markup pages 3.3-7, 3.3-8, and 3.3-50.

3. The term "calendar year" has been changed to "12 months" as requested by the NRC. This change affects ITS 3.3.2.1 page 3.3.2.1-2 and Bases page B 3.3.2.1-7, the Discussion of Changes (DOC) for ITS 3.3.2.1, DOC M.4 (page 3), the ISTS

markup page 3.3-16, the Justification for Deviations (JFD) to ITS 3.3.2.1, JFD 7 (page 1), and the ISTS Bases markup page B 3.3-49.

4. The Frequency for SR 3.3.2.1.5 has been changed from 24 months to 92 days, consistent with the actual trip setpoint methodology for the RBM channels. This change affects ITS 3.3.2.1 page 3.3.2.1-5 and Bases page B 3.3.2.1-11 and the ISTS markup pages 3.3-18 and Bases page B 3.3-52.

5. The Allowable Value for the RBM downscale has been modified to be consistent with the current setpoint analysis. The previous Allowable Value had been rounded up to the nearest tenth. This change affects ITS 3.3.2.1 page 3.3.2.1-6 and the ISTS markup page 3.3-20.

6. The Bases for ITS 3.3.2.1 has been modified to include the current fuel vendors RWM low power setpoint analysis and the DOC has been modified to reference the similar GE analysis, as requested by the NRC. This change affects ITS 3.3.2.1 Bases pages B 3.3.2.1-5, B 3.3.2.1-8, B 3.3.2.1-9, and B 3.3.2.1-14, the Discussion of Changes for ITS 3.3.2.1, DOC L.4 (page 7), and the ISTS Bases markup pages B 3.3-47, B 3.3-51, and B 3.3-55.

7. Markup errors have been corrected (the words for SR 3.3.2.1.5 and for ITS 3.3.3.1 Function 6 were inserted into the wrong locations, the words ", Level 2" in ITS 3.3.4.1 were circled but not lined out, the word "the" in Table 3.3.5.1-1 footnote a was not lined out, and the letter "S" was not lined out in the Applicability of ITS 3.3.8.2). These changes affect the ISTS 3.3.2.1 markup page 3.3-18, the ISTS 3.3.3.1 markup page 3.3-26, the ISTS 3.3.4.1 markup page 3.3-33, the ISTS 3.3.5.1 markup page 3.3-44, and the ISTS 3.3.8.2 markup page 3.3-78.

8. The Applicable Safety Analyses Bases for ITS 3.3.5.2, concerning the engineered safety feature discussion, have been modified to be consistent with similar words used in ITS 3.5.3. This change affects ITS 3.3.5.2 Bases page B 3.3.5.2-1 and the ISTS Bases markup page B 3.3-140.

9. Typographical errors have been corrected in the Bases of ITS 3.3.6.1 (footnote (c) has been changed to footnote (b) and Function 1.c has been changed to Function 1.d) and the Applicability of ITS 3.3.7.1 (addition of the word "and" in the first line). These changes affect ITS 3.3.6.1 Bases pages B 3.3.6.1-19 and B 3.3.6.1-22, the ISTS Bases markup pages B 3.3-174 and B 3.3-177, ITS 3.3.7.1 page 3.3.7.1-1, and the ISTS markup page 3.3-71.

10. An editorial correction has been made (the number of channels for Function 2.b has been changed from "2" to "1") to be consistent with the channel description in the Bases. This change affects ITS 3.3.8.1 page 3.3.8.1-3 and the ISTS markup page 3.3-77.

Section 3.4

1. The word "Identify" has been changed to "Verify," as requested by the NRC. In addition, the acronym "IGSCC" has been identified). These changes affect ITS

3.4.4 page 3.4.4-2 and Bases page B 3.4.4-4 and the ISTS markup page 3.4-8 and Bases page B 3.4-20.

2. The primary containment atmospheric particulate sampling system has been added back into the ITS as requested by the NRC. This change affects the Split Report, Summary Disposition Matrix page 6, Appendix A page 11 of 16, and Appendix B page 3 of 5, ITS 3.4.5 pages 3.4.5-1 and 3.4.5-2 and Bases pages B 3.4.5-1, B 3.4.5-2, B 3.4.5-3, B 3.4.5-4, and B 3.4.5-5, the CTS markup for ITS 3.4.5, pages 1 of 2 and 2 of 2, the Discussion of Changes for ITS 3.4.5, DOC M.1 (page 1), DOC M.2 (page 1), DOC LA.1 (page 2) and DOC R.1 (deleted from page 2), the ISTS markup pages 3.4-12, 3.4-13, 3.4-14, and 3.4-15, the Justification for Deviations to ITS 3.4.5, JFD 4 (page 1) and JFD 5 (page 1), and the ISTS Bases markup pages B 3.4-27, B 3.4-28, insert page B 3.4-28, B 3.4-29, B 3.4-30, insert page B 3.4-30, B 3.4-31, and B 3.4-32.

3. The statement concerning LCO Note 1 has been deleted from the Bases of ITS 3.4.8 ACTION A.1 since LCO Note 1 does not exempt OPERABILITY. This change affects ITS 3.4.8 Bases page B 3.4.8-3 and the ISTS Bases markup page B 3.4-44.

Section 3.5

1. The change committed to during discussion with the NRC to resolve a beyond scope issue related to the amount of water required in the contaminated condensate storage tanks has been made. This change affects ITS 3.5.2 page 3.5.2-3 and Bases pages B 3.5.2-1, B 3.5.2-4, and B 3.5.2-5, the CTS markup for ITS 3.5.2, pages 1 of 4, 2 of 4, and 3 of 4, the Discussion of Changes for ITS 3.5.2, DOC L.5 (deleted from page 7), the ISTS markup page 3.5-9 and Bases markup pages B 3.5-17 and B 3.5-20, and the No Significant Hazards Consideration, NSHC L.5 (deleted).

2. The Bases for ITS 3.5.2 have been modified to include the information relocated by ITS 3.5.2 DOC LA.1. This change affects ITS 3.5.2 Bases page B 3.5.2-1 and the ISTS Bases markup page B 3.5-17.

Section 3.6

1. The change committed to during discussions with the NRC to resolve a beyond scope issue related to the drywell-to-suppression chamber bypass leakage Surveillance has been made. This change affects ITS 3.6.1.1 page 3.6.1.1-2 and Bases pages B 3.6.1.1-4 and B 3.6.1.1-5, the CTS markup for ITS 3.6.1.1, pages 2 of 3 and 3 of 3, the Discussion of Changes for ITS 3.6.1.1, DOC L.3 (deleted from page 4), the ISTS markup page 3.6-2, the Justification for Deviations to ITS 3.6.1.1, JFD 2 (page 1), the ISTS Bases markup page B 3.6-4, insert page B 3.6-4 (deleted), and B 3.6-5, and the No Significant Hazards Consideration for ITS 3.6.1.1, NSHC L.3 (deleted from pages 3 and 4).

2. The Frequency for SR 3.6.2.4.2 has been changed from 5 years to 10 years, as requested by the NRC. In addition, an editorial correction to the ISTS Bases markup has also been made (the word "RHR" has been deleted from the Bases header). These changes affect ITS 3.6.2.4 page 3.6.2.4-2 and Bases page B

3.6.2.4-4, the Discussion of Changes for ITS 3.6.2.4, DOC M.1 (page 1), the ISTS markup page 3.6-38, the Justification for Deviations to ITS 3.6.2.4, JFD 5 (page 1), and the ISTS Bases markup page B 3.6-74.

3. The change committed to during discussions with the NRC to resolve RAI 3.6.4.1-1 has been made. This change affects ITS 3.6.4.1 page 3.6.4.1-2 and Bases pages B 3.6.4.1-4 and B 3.6.4.1-5, the CTS markup for ITS 3.6.4.1, page 1 of 1, the Discussion of Changes for ITS 3.6.4.1, DOC M.2 (page 2), the ISTS markup page 3.6-48, the Justification for Deviations to ITS 3.6.4.1, JFD 2 (page 1), and the ISTS Bases markup page B 3.6-100.

Section 3.8

1. A typographical error in Note 1 of the Surveillance Requirements has been corrected (the term "SR" has been added before the second SR number). This change affects ITS 3.8.1 page 3.8.1-6 and the ISTS markup insert page 3.8-6.

2. Two ISTS markup errors in ITS 3.8.1 have been corrected (the proper voltage has been provided in SR 3.8.1.9.a and the redundant unit "V" has been deleted from SR 3.8.1.13.a). This change affects the ISTS 3.8.1 markup pages 3.8-8 and 3.8-11.

3. The word "and" has been deleted in LCO 3.8.4.a, consistent with the format of the ITS. This change affects ITS 3.8.4 page 3.8.4-1 and the ISTS markup insert page 3.8-24a.

4. An ISTS markup error has been corrected in SR 3.8.4.1 (the redundant unit "V" has been deleted). This change affects the ISTS markup page 3.8-25.

5. ITS LCO 3.8.5 has been modified as requested by the NRC. In addition, a typographical error has been corrected. These changes affect ITS 3.8.5 page 3.8.5-1 and Bases page B 3.8.5-2, the CTS markup for ITS 3.8.5, page 1 of 1, the Discussion of Changes for ITS 3.8.5, DOC M.1 (page 1), the ISTS markup page 3.8-28, the Justification for Deviations to ITS 3.8.5, JFD 7 (page 1), and the ISTS Bases markup page B 3.8-61.

6. ITS LCO 3.8.7 Condition C has been modified to be consistent with the LaSalle ITS. This change affects ITS 3.8.7 page 3.8.7-2 and Bases page B 3.8.7-9 and the ISTS markup insert page 3.8-38 and Bases insert page B 3.8-86.

Section 3.9

1. A change was made to the JFD as requested by the NRC. This change affects the Justification for Deviations to ITS 3.9.1, JFD 3 (page 1).

Section 3.10

1. Typographical errors in the Background Section of the Bases has been corrected (the word "scram" has been changed to "scrams" in three locations). This change affects ITS 3.10.1 Bases page B 3.10.1-1 and the ISTS Bases markup page B 3.10-6.

2. A typographical error in NUREG-1433 has been corrected to be consistent with NUREG-1434 (the word "and" has been added). This change affects ITS 3.10.3 page 3.10.3-1 and the ISTS 3.10.3 markup page 3.10-9.

Chapter 5.0

1. The change committed to during discussions with the NRC to resolve RAI 5.0-1 has been made. The change affects ITS 5.1 page 5.1-1, the Discussion of Changes for ITS 5.1, DOC M.1 (page 1) and DOC LA.2 (page 1), the ISTS markup page 5.0-1, and the Justification for Deviations to ITS 5.1, JFD 3 (page1) and JFD 4 (page 1).

2. A typographical error has been corrected in ITS 5.2.2.b, in the Justification for Deviations to ITS 5.2, and in the Discussion of Changes for ITS 5.3 (the reference to Specification 5.2.2.g has been changed to 5.2.2.f). In addition, the reference to 5.5.2.a in ITS 5.2 JFD 6 has been changed to 5.2.2.a. These changes affect ITS 5.2 page 5.2-2, the ISTS markup page 5.0-3, the Justification for Deviations to ITS 5.2, JFD 6 (page 1), and the Discussion of Changes for ITS 5.3, DOC A.2 (page 1).

3. Markup errors in ITS 5.6 and ITS 5.7 have been corrected (the term "mrems" has been changed to "mrem" and the word "documents" has been changed to "document"). These changes affect the ISTS 5.6 markup insert page 5.0-18 and the ISTS 5.7 markup page 5.0-23.

Dresden ITS Rev. D Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 1	
SPLIT REPORT, CHAPTERS 1.0 AND 2.0, AND SECTION 3.0	
DISCARD	**INSERT**
Split Report, Summary Disposition Matrix for Dresden 2 and 3, Page 6	Split Report, Summary Disposition Matrix for Dresden 2 and 3, Page 6
Split Report, Appendix A, Page 11 of 16	Split Report, Appendix A, Page 11 of 16
Split Report, Appendix A, Page 3 of 5	Split Report, Appendix A, Page 3 of 5
ITS page 1.1-3	ITS page 1.1-3
ISTS markup page 1.1-3	ISTS markup page 1.1-3
ISTS markup page 1.4-5	ISTS markup page 1.4-5

Dresden ITS Rev. D Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 2	
SECTIONS 3.1 AND 3.2	
DISCARD	**INSERT**
ISTS markup page 3.1-21	ISTS markup page 3.1-21
ITS Bases pages B 3.2.4-1 through B 3.2.4-6	ITS Bases pages B 3.2.4-1 through B 3.2.4-6
ISTS Bases markup insert page B 3.2-14	ISTS Bases markup insert page B 3.2-14

Dresden ITS Rev. D Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 3	
SECTION 3.3	
DISCARD	**INSERT**
ITS page 3.3.1.1-1	ITS page 3.3.1.1-1
ITS pages 3.3.1.1-7 through 3.3.1.1-9	ITS pages 3.3.1.1-7 through 3.3.1.1-9
ITS page 3.3.2.1-2	ITS page 3.3.2.1-2
ITS pages 3.3.2.1-5 and 3.3.2.1-6	ITS pages 3.3.2.1-5 and 3.3.2.1-6
ITS page 3.3.5.2-2	ITS page 3.3.5.2-2
ITS page 3.3.7.1-1	ITS page 3.3.7.1-1
ITS page 3.3.8.1-3	ITS page 3.3.8.1-3
ITS Bases page B 3.3.2.1-5	ITS Bases page B 3.3.2.1-5
ITS Bases pages B 3.3.2.1-7 through B 3.3.2.1-9	ITS Bases pages B 3.3.2.1-7 through B 3.3.2.1-9
ITS Bases page B 3.3.2.1-11	ITS Bases page B 3.3.2.1-11
ITS Bases page B 3.3.2.1-14	ITS Bases page B 3.3.2.1-14
ITS Bases page B 3.3.5.2-1	ITS Bases page B 3.3.5.2-1
ITS Bases page B 3.3.6.1-19	ITS Bases page B 3.3.6.1-19
ITS Bases page B 3.3.6.1-22	ITS Bases page B 3.3.6.1-22
Discussion of Changes for ITS 3.3.2.1 page 3	Discussion of Changes for ITS 3.3.2.1 page 3
Discussion of Changes for ITS 3.3.2.1 Pages 7 and 8	Discussion of Changes for ITS 3.3.2.1 Pages 7 and 8

VOLUME 4	
SECTION 3.3	
DISCARD	**INSERT**
ISTS markup page 3.3-1	ISTS markup page 3.3-1
ISTS markup pages 3.3-5 through 3.3-8	ISTS markup pages 3.3-5 through 3.3-8
ISTS markup page 3.3-16	ISTS markup page 3.3-16
ISTS markup page 3.3-18	ISTS markup page 3.3-18
ISTS markup page 3.3-20	ISTS markup page 3.3-20
Justification for Deviations to ITS 3.3.2.1 page 1	Justification for Deviations to ITS 3.3.2.1 page 1
ISTS markup page 3.3-26	ISTS markup page 3.3-26
ISTS markup page 3.3-33	ISTS markup page 3.3-33
ISTS markup page 3.3-44	ISTS markup page 3.3-44
ISTS markup page 3.3-50	ISTS markup page 3.3-50
ISTS markup page 3.3-71	ISTS markup page 3.3-71
ISTS markup page 3.3-77	ISTS markup page 3.3-77
ISTS markup page 3.3-78	ISTS markup page 3.3-78
ISTS Bases markup page B 3.3-47	ISTS Bases markup page B 3.3-47
ISTS Bases markup page B 3.3-49	ISTS Bases markup page B 3.3-49
ISTS Bases markup page B 3.3-51	ISTS Bases markup page B 3.3-51
ISTS Bases markup page B 3.3-52	ISTS Bases markup page B 3.3-52
ISTS Bases markup page B 3.3-55	ISTS Bases markup page B 3.3-55
ISTS Bases markup page B 3.3-140	ISTS Bases markup page B 3.3-140
ISTS Bases markup page B 3.3-174	ISTS Bases markup page B 3.3-174
ISTS Bases markup page B 3.3-177	ISTS Bases markup page B 3.3-177

VOLUME 5	
SECTIONS 3.4 AND 3.5	
DISCARD	**INSERT**
ITS page 3.4.4-2	ITS page 3.4.4-2
ITS pages 3.4.5-1 and 3.4.5-2	ITS pages 3.4.5-1 and 3.4.5-2
ITS Bases page B 3.4.4-4	ITS Bases page B 3.4.4-4
ITS Bases pages B 3.4.5-1 through B 3.4.5-4	ITS Bases pages B 3.4.5-1 through B 3.4.5-5
ITS Bases pages B 3.4.8-3 and B 3.4.8-4	ITS Bases pages B 3.4.8-3 and B 3.4.8-4
CTS markup for ITS 3.4.5 pages 1of 2 and 2 of 2	CTS markup for ITS 3.4.5 pages 1of 2 and 2 of 2
Discussion of Changes for ITS 3.4.5 pages 1 and 2	Discussion of Changes for ITS 3.4.5 pages 1 and 2
ISTS markup page 3.4-8	ISTS markup page 3.4-8
ISTS markup pages 3.4-12 through 3.4-15	ISTS markup pages 3.4-12 through 3.4-15
Justification for Deviations to ITS 3.4.5 page 1	Justification for Deviations to ITS 3.4.5 page 1
ISTS Bases markup page B 3.4-20	ISTS Bases markup page B 3.4-20
ISTS Bases markup page B 3.4-27	ISTS Bases markup page B 3.4-27
ISTS Bases markup page B 3.4-28 and insert page B 3.4-28	ISTS Bases markup page B 3.4-28 and insert page B 3.4-28
ISTS Bases markup page B 3.4-29	ISTS Bases markup page B 3.4-29
ISTS Bases markup page B 3.4-30	ISTS Bases markup page B 3.4-30 and insert page B 3.4-30
ISTS Bases markup pages B 3.4-31 and B 3.4-32	ISTS Bases markup pages B 3.4-31 and B 3.4-32
ISTS Bases markup page B 3.4-44	ISTS Bases markup page B 3.4-44
ITS page 3.5.2-3	ITS page 3.5.2-3
ITS Bases page B 3.5.2-1	ITS Bases page B 3.5.2-1
ITS Bases pages B 3.5.2-4 and B 3.5.2-5	ITS Bases pages B 3.5.2-4 and B 3.5.2-5
CTS markup for ITS 3.5.2 pages 1 of 4 through 3 of 4	CTS markup for ITS 3.5.2 pages 1 of 4 through 3 of 4
Discussion of Changes for ITS 3.5.2 page 7	Discussion of Changes for ITS 3.5.2 page 7
ISTS markup page 3.5-9	ISTS markup page 3.5-9
ISTS Bases markup page B 3.5-17	ISTS Bases markup page B 3.5-17
ISTS Bases markup page B 3.5-20	ISTS Bases markup page B 3.5-20
No Significant Hazards Consideration for ITS 3.5.2 page 5	None

VOLUME 6	
SECTION 3.6	
DISCARD	**INSERT**
ITS page 3.6.1.1-2	ITS page 3.6.1.1-2
ITS page 3.6.2.4-2	ITS page 3.6.2.4-2
ITS page 3.6.4.1-2	ITS page 3.6.4.1-2
ITS Bases pages B 3.6.1.1-4 and B 3.6.1.1-5	ITS Bases pages B 3.6.1.1-4 and B 3.6.1.1-5
ITS Bases page B 3.6.2.4-4	ITS Bases page B 3.6.2.4-4
ITS Bases pages B 3.6.4.1-4 and B 3.6.4.1-5	ITS Bases pages B 3.6.4.1-4 and B 3.6.4.1-5
CTS markup for ITS 3.6.1.1 pages 2 of 3 and 3 of 3	CTS markup for ITS 3.6.1.1 pages 2 of 3 and 3 of 3
Discussion of Changes for ITS 3.6.1.1 pages 4 and 5	Discussion of Changes for ITS 3.6.1.1 page 4
Discussion of Changes for ITS 3.6.2.4 page 1	Discussion of Changes for ITS 3.6.2.4 page 1
CTS markup for ITS 3.6.4.1 page 1 of 1	CTS markup for ITS 3.6.4.1 page 1 of 1
Discussion of Changes for ITS 3.6.4.1 Pages 2 and 3	Discussion of Changes for ITS 3.6.4.1 Pages 2 and 3

VOLUME 7	
SECTION 3.6	
DISCARD	**INSERT**
ISTS markup page 3.6-2	ISTS markup page 3.6-2
Justification for Deviations to ITS 3.6.1.1 page 1	Justification for Deviations to ITS 3.6.1.1 page 1
ISTS markup page 3.6-38	ISTS markup page 3.6-38
Justification for Deviations to ITS 3.6.2.4 page 1	Justification for Deviations to ITS 3.6.2.4 page 1
ISTS markup page 3.6-48	ISTS markup page 3.6-48
Justification for Deviations to ITS 3.6.4.1 page 1	Justification for Deviations to ITS 3.6.4.1 page 1
ISTS Bases markup page B 3.6-4 and insert page B 3.6-4	ISTS Bases markup page B 3.6-4
ISTS Bases markup page B 3.6-5	ISTS Bases markup page B 3.6-5
ISTS Bases markup page B 3.6-74	ISTS Bases markup page B 3.6-74
ISTS Bases markup page B 3.6-100	ISTS Bases markup page B 3.6-100
No Significant Hazards Consideration for ITS 3.6.1.1 pages 3 and 4	None

VOLUME 9	
SECTION 3.8	
DISCARD	**INSERT**
ITS page 3.8.1-6	ITS page 3.8.1-6
ITS page 3.8.4-1	ITS page 3.8.4-1
ITS page 3.8.5-1	ITS page 3.8.5-1
ITS page 3.8.7-2	ITS page 3.8.7-2
ITS Bases pages B 3.8.5-2 and B 3.8.5-3	ITS Bases pages B 3.8.5-2 and B 3.8.5-3
ITS Bases page B 3.8.7-9	ITS Bases page B 3.8.7-9
CTS markup for ITS 3.8.5 page 1 of 1	CTS markup for ITS 3.8.5 page 1 of 1
Discussion of Changes for ITS 3.8.5 pages 1 through 3	Discussion of Changes for ITS 3.8.5 pages 1 through 3
ISTS markup insert page 3.8-6	ISTS markup insert page 3.8-6
ISTS markup page 3.8-8	ISTS markup page 3.8-8
ISTS markup page 3.8-11	ISTS markup page 3.8-11
ISTS markup insert page 3.8-24a	ISTS markup insert page 3.8-24a
ISTS markup page 3.8-25	ISTS markup page 3.8-25
ISTS markup page 3.8-28	ISTS markup page 3.8-28
Justification for Deviations to ITS 3.8.5 page 1	Justification for Deviations to ITS 3.8.5 page 1
ISTS markup insert page 3.8-38	ISTS markup insert page 3.8-38
ISTS Bases markup page B 3.8-61	ISTS Bases markup page B 3.8-61
ISTS Bases markup insert page B 3.8-86	ISTS Bases markup insert page B 3.8-86

VOLUME 10	
SECTIONS 3.9 AND 3.10	
DISCARD	**INSERT**
Justification for Deviations to ITS 3.9.1 page 1	Justification for Deviations to ITS 3.9.1 page 1
ITS page 3.10.3-1	ITS page 3.10.3-1
ITS Bases page B 3.10.1-1	ITS Bases page B 3.10.1-1
ISTS markup page 3.10-9	ISTS markup page 3.10-9
ISTS Bases markup page B 3.10-6	ISTS Bases markup page B 3.10-6

VOLUME 11	
CHAPTERS 4.0 AND 5.0	
DISCARD	**INSERT**
ITS page 5.1-1	ITS page 5.1-1
ITS page 5.2-2	ITS page 5.2-2
Discussion of Changes for ITS 5.1 pages 1 and 2	Discussion of Changes for ITS 5.1 pages 1 and 2
Discussion of Changes for ITS 5.3 page 1	Discussion of Changes for ITS 5.3 page 1
ISTS markup page 5.0-1	ISTS markup page 5.0-1
Justification for Deviations to ITS 5.1 page 1	Justification for Deviations to ITS 5.1 page 1
ISTS markup page 5.0-3	ISTS markup page 5.0-3
Justification for Deviations to ITS 5.2 page 1	Justification for Deviations to ITS 5.2 page 1
ISTS markup insert page 5.0-18	ISTS markup insert page 5.0-18
ISTS markup page 5.0-23	ISTS markup page 5.0-23

SUMMARY DISPOSITION MATRIX FOR DRESDEN 2 AND 3

CTS NUMBER	TITLE	ITS NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION[a]
3/4.6	PRIMARY SYSTEM BOUNDARY			
3/4.6.A	Recirculation Loops	3.4.1	Yes-2	Recirculation loop flow is an initial condition in the safety analysis.
3/4.6.B	Jet Pumps	3.4.2	Yes-3	Jet pump operability is assumed in the LOCA analysis to assure adequate core reflood capability.
3/4.6.C	Recirculation Pumps	3.4.1	Yes-2	Recirculation loop flow (pump speed) mismatch, within limits, is an initial condition in the safety analysis.
3/4.6.D	Idle Recirculation Loop Startup	3.4.9	Yes-2	Establishes initial conditions to operation such that operation is prohibited in areas or at temperature rate changes that might cause undetected flaws to propagate, in turn challenging the reactor coolant pressure boundary integrity.
3/4.6.E	Safety Valves	3.4.3	Yes-3	A minimum number of safety valves is assumed in the safety analyses to mitigate overpressure events.
3/4.6.F	Relief Valves	3.3.6.3 3.4.3 3.6.1.6	Yes-3	A minimum number of relief valves is assumed in the transient and containment loading safety analysis.
3/4.6.G	Leakage Detection Systems	3.4.5	Yes	The drywell floor drain sump leak detection instrumentation is used to indicate a significant abnormal condition of the reactor coolant system pressure boundary. The primary containment atmospheric particulate radioactivity sampling system is being maintained to be consistent with NUREG-1433.
3/4.6.H	Operational Leakage	3.4.4	Yes-2	Leakage beyond limits would indicate an abnormal condition of the reactor coolant system pressure boundary. Operation in this condition is unanalyzed and may result in reactor coolant system pressure boundary failure.
3/4.6.I	Relocated by Amendment Nos. 173 (Unit 2) and 169 (Unit 3)			
3/4.6.J	Specific Activity	3.4.6	Yes-2	Specific activity provides an indication of the onset of significant fuel cladding failure and is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment.

(a) The applicable safety analyses are discussed in the Bases for the individual Technical Specifications.

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1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	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;" Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity." D
FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC)	The FDLRC shall be 1.2 times the LHGR existing at a given location divided by the product of the transient LHGR limit and the fraction of RTP.
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; b. <u>Unidentified LEAKAGE</u> All LEAKAGE into the drywell that is not identified LEAKAGE; c. <u>Total LEAKAGE</u> Sum of the identified and unidentified LEAKAGE; and d. <u>Pressure Boundary LEAKAGE</u> LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

(continued)

<CTS>

<1.0> 1.1 Definitions

<L.2>

DOSE EQUIVALENT I-131 (continued)	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 or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity".
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC RPT) SYSTEM RESPONSE TIME	The EOC RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by [the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint] to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured, [except for the breaker arc suppression time, which is not measured but is validated to conform to the manufacturer's design value].
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential,

<A.1> INSERT 1

(continued)

<CTS>

<A.1B> 1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

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

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

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

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.4.A.1.b>	SR 3.1.7.1	Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1, or ≥ 4530 gallons.	24 hours
<4.4.A.1.a>	SR 3.1.7.2	Verify temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-2.	24 hours
<4.4.A.1.c>	SR 3.1.7.3	Verify temperature of pump suction piping is within the limits of Figure 3.1.7-2.	24 hours ≥ 83° F
<4.4.A.2.a>	SR 3.1.7.4	Verify continuity of explosive charge.	31 days
<4.4.A.2.b> <4.4.A Footnote (a)>	SR 3.1.7.5	Verify the concentration of ~~boron~~ sodium pentaborate in solution is within the limits of Figure 3.1.7-1.	31 days AND Once within 24 hours after water or ~~boron~~ is added to solution AND Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2

2

3

D

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable final design criteria are discussed in UFSAR, Sections 3.1.2.2.1, 3.1.2.2.4, 3.1.2.3.1, and 3.1.2.3.10 (Ref. 1). This LCO is provided to require the APRM gain or APRM Flow Biased Neutron Flux—High Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b) to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

The condition of excessive power peaking is determined by Fuel Design Limit Ratio for Centerline Melt (FDLRC), which is defined as:

$$FDLRC = \frac{(LHGR)(1.2)}{(TLHGR)(FRTP)} ;$$

where LHGR is the Linear Heat Generation Rate, FRTP is the Fraction of Rated Thermal Power determined by dividing the measured THERMAL POWER by the RTP, and TLHGR is the Transient Linear Heat Generation Rate limit. The TLHGR limit is specified in the COLR and shall be the limit that protects against fuel centerline melting and the fuel cladding 1% plastic strain during transient conditions throughout the life of the fuel. D

Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during AOOs beginning at any power level and terminating at 120% RTP (APRM Fixed Neutron Flux—High Allowable Value). The APRM Flow Biased Neutron Flux—High Function Allowable Value must be adjusted to ensure that the TLHGR limit is not violated for any power distribution. When FDLRC is greater than 1.0, excessive power peaking exists. To maintain margins similar to those at RTP conditions, the APRM Flow Biased Allowable Value is decreased by 1/FDLRC. As an alternative, this adjustment may also be accomplished by increasing the gain of the APRM by FDLRC. Increasing the APRM gain raises the initial APRM

(continued)

BASES

BACKGROUND (continued)

reading closer to the Flow Biased Allowable Value such that a scram would be received at the same point in a transient as if the Allowable Value had been reduced. Thus, increasing the APRM gain by FDLRC provides the same degree of protection as reducing the APRM Flow Biased Neutron Flux—High Function Allowable Value by 1/FDLRC. Either of these adjustments has effectively the same result as maintaining FDLRC less than or equal to 1.0, and thus, maintains RTP margins for APLHGR, MCPR, and LHGR.

The normally selected APRM Flow Biased Neutron Flux—High Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM Allowable Value is supported by the analyses presented in Reference 3 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Flow Biased Neutron Flux—High Function Allowable Value may be reduced during operation when FDLRC indicates an excessive power peaking distribution.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR, MCPR, and LHGR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

UFSAR safety analyses (Ref. 2) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, MCPR, and LHGR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR),"

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of the APLHGR, the MCPR, or the LHGR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the fuel design limits could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM Flow Biased Neutron Flux—High Function Allowable Value is adjusted downward by 1/FDLRC, or the APRM gain is adjusted upward by FDLRC. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the APRM Flow Biased Neutron Flux—High Function Allowable Value, dependent on the increased peaking that may be encountered.

The APRM gain and setpoint satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM Flow Biased Neutron Flux—High Function Allowable Value by multiplying the APRM Flow Biased Neutron Flux—High Function Allowable Value by 1/FDLRC; or

c. Increasing APRM gains to cause the APRM to read greater than or equal to 100% times FRTP times FDLRC. This condition is to account for the reduction in

(continued)

BASES

LCO (continued)

margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during AOOs beginning at any power level and terminating at 120% of RTP. When FDLRC is greater than 1.0, excessive power peaking exists. To compensate for this condition, the APRM Flow Biased Neutron Flux—High Function Allowable Value is adjusted downward by 1/FDLRC or the APRM gain is adjusted upward by FDLRC. When the reactor is operating with the peaking less than the design value, it is not necessary to modify the APRM Flow Biased Neutron Flux—High Function Allowable Value. Modifying the APRM Flow Biased Allowable Value or adjusting the APRM gain is equivalent to maintaining FDLRC less than or equal to 1.0, as stated in the LCO.

For compliance with LCO 3.2.4.b (APRM Flow Biased Neutron Flux—High Function Allowable Value modification) or LCO 3.2.4.c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, Function 2.b are required to be modified or adjusted. In addition, each APRM may be allowed to have its gain adjusted or Allowable Value modified independently of other APRMs that are having their gain adjusted or Allowable Value modified.

APPLICABILITY

The FDLRC limit, APRM gain adjustment, or APRM Flow Biased Neutron Flux—High Function Allowable Value is provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3 sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If the APRM gain or Flow Biased Neutron Flux—High Function Allowable Value is not within limits while FDLRC has exceeded 1.0, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit may be reduced. Therefore, prompt action should be taken to

(continued)

BASES

ACTIONS

A.1 (continued)

restore the FDLRC to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the FDLRC to within limits or to adjust the APRM gain or modify the APRM Flow Biased Neutron Flux – High Function Allowable Value to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

If FDLRC, the APRM gain or Flow Biased Neutron Flux – High Function Allowable Value cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The FDLRC is required to be calculated and compared to 1.0 or APRM gain adjusted or APRM Flow Biased Neutron Flux – High Function Allowable Value modified to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the FDLRC and, assuming FDLRC is greater than 1.0, the appropriate APRM gain or APRM Flow Biased Neutron Flux – High Function Allowable Value, and are not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or Flow Biased Neutron Flux – High Function circuitry. SR 3.2.4.1 and SR 3.2.4.2 have been modified by Notes, which clarify that the respective SR does not have to be met if the alternate requirement demonstrated by the other SR is satisfied. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1), MCPR (LCO 3.2.2), and LHGR

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

(LCO 3.2.3). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER $\geq$ 25% RTP is achieved is acceptable given the large inherent margin to APLHGR, MCPR, and LHGR operating limits at low power levels.

The 12 hour Frequency of SR 3.2.4.2 is required when FDLRC is greater than 1.0, because more rapid changes in power distribution are typically expected.

REFERENCES

1. UFSAR, Sections 3.1.2.2.1, 3.1.2.2.4, 3.1.2.3.1, and 3.1.2.3.10.

2. UFSAR, Chapter 15.

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The condition of excessive power peaking is determined by using the Fuel Design Limit Ratio for Centerline Melt (FDLRC), which is defined as:

$$FDLRC = \frac{(LHGR)(1.2)}{(TLHGR)(FRTP)} ;$$

where LHGR is the Linear Heat Generation Rate, FRTP is the Fraction of Rated Thermal Power determined by dividing the measured THERMAL POWER by the RTP, and TLHGR is the Transient Linear Heat Generation Rate limit. The TLHGR limit is specified in the COLR and shall be the limit that protects against fuel centerline melting and the fuel cladding 1% plastic strain during transient conditions throughout the life of the fuel.

Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during AOOs beginning at any power level and terminating at 120% RTP (APRM Fixed Neutron Flux—High Allowable Value). The APRM Flow Biased Neutron Flux—High Function Allowable Value must be adjusted to ensure that the TLHGR limit is not violated for any power distribution. When FDLRC is greater than 1.0, excessive power peaking exists. To maintain margins similar to those at RTP conditions, the APRM Flow Biased Allowable Value is decreased by 1/FDLRC. As an alternative, this adjustment may also be accomplished by increasing the gain of the APRM by FDLRC. Increasing the APRM gain raises the initial APRM reading closer to the Flow Biased Allowable Value such that a scram would be received at the same point in a transient as if the Allowable Value had been reduced. Thus, increasing the APRM gain by FDLRC provides the same degree of protection as reducing the APRM Flow Biased Neutron Flux—High Function Allowable Value by 1/FDLRC. Either of these adjustments has effectively the same result as maintaining FDLRC less than or equal to 1.0, and thus, maintains RTP margins for APLHGR, MCPR, and LHGR.

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

----------NOTES----------

1. Separate Condition entry is allowed for each channel.
2. When Functions 2.b and 2.c channels are inoperable due to APRM indication not within limits, entry into associated Conditions and Required Actions may be delayed for up to 2 hours if the APRM is indicating a lower power value than the calculated power, and for up to 12 hours if the APRM is indicating a higher power value than the calculated power. D C

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
	OR		
	A.2	Place associated trip system in trip.	12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	OR		
	B.2	Place one trip system in trip.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.17	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Function 1.a, not required to be performed when entering MODE 2 from MODE 1 until 24 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.18	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.19	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Function 5 "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. ---------- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

1D

1D

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 121/125 divisions of full scale
	5[a]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 121/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.18	NA
	5[a]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.18	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.18	≤ 17.1% RTP
b. Flow Biased Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 0.58 W + 63.5% RTP and ≤ 120% RTP[b]

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.58 W + 59.2% and ≤ 118.5% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	
2. Average Power Range Monitors (continued)						
c. Fixed Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 120% RTP	A D
d. Inop	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.18	NA	
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1058 psig	A D
4. Reactor Vessel Water Level – Low	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ 10.24 inches	A
5. Main Steam Isolation Valve – Closure	1, 2[c]	8	F	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 9.5% closed	B
6. Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1.94 psig	A

(continued)

(c) With reactor pressure ≥ 600 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1.1 Verify ≥ 12 rods withdrawn. OR C.2.1.2 Verify by administrative methods that startup with RWM inoperable has not been performed in the last 12 months. AND C.2.2 Verify movement of control rods is in compliance with the analyzed rod position sequence by a second licensed operator or other qualified member of the technical staff.	Immediately Immediately During control rod movement
D. RWM inoperable during reactor shutdown.	D.1 Verify movement of control rods is in compliance with analyzed rod position sequence by a second licensed operator or other qualified member of the technical staff.	During control rod movement

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.5	----------NOTE---------- Neutron detectors are excluded. ---------- Verify the RBM is not bypassed when THERMAL POWER is ≥ 30% RTP and when a peripheral control rod is not selected.	92 days
SR 3.3.2.1.6	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	24 months
SR 3.3.2.1.7	----------NOTE---------- Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. ---------- Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with analyzed rod position sequence.	Prior to declaring RWM OPERABLE following loading of sequence into RWM
SR 3.3.2.1.9	Verify the bypassing and position of control rods required to be bypassed in RWM by a second licensed operator or other qualified member of the technical staff.	Prior to and during the movement of control rods bypassed in RWM

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	As specified in the COLR
b. Inop	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.5	NA
c. Downscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	≥ 4.03% RTP A D
2. Rod Worth Minimizer	1[(b)],2[(b)]	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.6 SR 3.3.2.1.8 SR 3.3.2.1.9	NA
3. Reactor Mode Switch – Shutdown Position	(c)	2	SR 3.3.2.1.7	NA

(a) THERMAL POWER ≥ 30% RTP and no peripheral control rod selected.

(b) With THERMAL POWER ≤ 10% RTP.

(c) Reactor mode switch in the shutdown position.

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 Reactor Vessel Pressure – High Function maintains IC initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.5.2.1	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.5.2.2	----------NOTE---------- Not required for the time delay portion of the channel. ---------- Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 1068 psig.	 92 days
SR 3.3.5.2.3	Perform CHANNEL CALIBRATION for the time delay portion of the channel. The Allowable Value shall be ≤ 17 seconds.	24 months
SR 3.3.5.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

LCO 3.3.7.1 Two channels of the Reactor Building Ventilation System – High High Radiation Alarm Function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
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	Declare CREV System inoperable.	1 hour from discovery of loss of CREV System Instrumentation alarm capability in both trip systems
	AND		
	A.2	Restore channel to OPERABLE status.	6 hours

(continued)

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER BUS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. 4160 V Essential Service System Bus Undervoltage (Loss of Voltage)	2	SR 3.3.8.1.3 SR 3.3.8.1.4 SR 3.3.8.1.5	≥ 2796.85 V and ≤ 3063.20 V
2. 4160 V Essential Service System Bus Undervoltage (Degraded Voltage)			
a. Bus Undervoltage/Time Delay	2	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.5	≥ 3861 V and ≤ 3911 V with time delay ≥ 5.7 seconds and ≤ 8.3 seconds
b. Time Delay (No LOCA)	1	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.5	≥ 279 seconds and ≤ 321 seconds

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Rod Worth Minimizer (continued)

the RWM is available and required to be OPERABLE (Ref. 9). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the analyzed rod position sequence. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the analyzed rod position sequence, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is $\leq$ 10% RTP. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA (Refs. 4, 9, 10, and 11). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical. D

3. Reactor Mode Switch – Shutdown Position

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch – Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

The Reactor Mode Switch – Shutdown Position Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Two channels are required to be OPERABLE to ensure that no single channel failure will preclude a rod block when required. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on reactor mode switch position.

(continued)

BASES

ACTIONS (continued)

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM during withdrawal of one or more of the first 12 control rods was not performed in the last 12 months. These requirements minimize the number of reactor startups initiated with the RWM inoperable. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2. Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other task qualified member of the technical staff (e.g., shift technical advisor or reactor engineer).

The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

D.1

With the RWM inoperable during a reactor shutdown, the operator is still capable of enforcing the prescribed control rod sequence. Required Action D.1 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other task qualified member of the technical

(continued)

BASES

ACTIONS

D.1 (continued)

staff (e.g., shift technical advisor or reactor engineer). The RWM may be bypassed under these conditions to allow the reactor shutdown to continue.

E.1 and E.2

With one Reactor Mode Switch – Shutdown Position control rod withdrawal block channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. However, since the Required Actions are consistent with the normal action of an OPERABLE Reactor Mode Switch – Shutdown Position Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SDM ensured by LCO 3.1.1. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note to indicate that when an RBM 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 control rod block 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 analysis (Ref. 12) D

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control "Relay Select Marix" System input. 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. C

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 92 days is based on reliability analyses (Ref. 13). D

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system 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. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs and by attempting to select a control rod not in compliance with C

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.5

The RBM is automatically bypassed when power is below a specified value or if a peripheral control rod is selected. The power level is determined from the APRM signals input to each RBM channel. The automatic bypass setpoint must be verified periodically to be < 30% RTP. In addition, it must also be verified that the RBM is not bypassed when a control rod that is not a peripheral control rod is selected (only one non-peripheral control rod is required to be verified). If any bypass setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the APRM channel can be placed in the conservative condition to enable the RBM. If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.9. The 92 day Frequency is based on the actual trip setpoint methodology utilized for these channels. D

SR 3.3.2.1.6

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass setpoint must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

SR 3.3.2.1.7

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch – Shutdown Position Function to ensure that the entire 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 C

(continued)

BASES

REFERENCES (continued)

9. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

10. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

11. EMF-2237(P), "Dresden Units 2 and 3 Reduced Low Power Set Point Analysis for Control Rod Drop Accident," July 1999. D

12. GENE-770-06-1-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992. D

13. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988. D B

B 3.3 INSTRUMENTATION

B 3.3.5.2 Isolation Condenser (IC) System Instrumentation

BASES

BACKGROUND

The purpose of the IC System instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser). A more complete discussion of IC System operation is provided in the Bases of LCO 3.5.3, "IC System."

The IC System may be initiated by either automatic or manual means. Automatic initiation occurs for sustained (about 17 seconds) conditions of reactor vessel pressure high. The variable is monitored by four pressure switches that are connected to four time delay relays. The outputs of the time delay relays are connected in a one-out-of-two logic to a trip relay. The output of the trip relays are connected in a two-out-of-two logic arrangement. Once initiated, the IC logic can be overridden by the operator.

APPLICABLE SAFETY ANALYSES

The function of the IC System to provide core cooling to the reactor is used to respond to a main steam line isolation event. Although the IC System is an Engineered Safety Feature System, no credit is taken in the accident analyses for IC System operation. Based on its contribution to the reduction of overall plant risk, however, the IC System, and therefore its instrumentation, satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). D

LCO

The OPERABILITY of the IC System instrumentation is dependent upon the OPERABILITY of the four channels of the Reactor Vessel Pressure—High Function. Each channel must have its setpoint within the Allowable Value specified in SR 3.3.5.2.2. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

The Allowable Value for the IC System instrumentation Function is specified in the SR. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6.b. Reactor Vessel Water Level – Low (continued)

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 (two channels per trip system) of the 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. As noted (footnote (b) to Table 3.3.6.1-1), only one channel per trip system (with an isolation signal available to one shutdown cooling pump suction isolation valve) of the Reactor Vessel Water Level–Low Function is required to be OPERABLE in MODES 4 and 5, provided the Shutdown Cooling System integrity is maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system. 1 D

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 the capability to cool the fuel may be threatened.

The Reactor Vessel Water Level – Low Function is only required to be OPERABLE in MODES 3, 4, and 5 to prevent this potential flow path from lowering the reactor vessel level to the top of the fuel. In MODES 1 and 2, another isolation (i.e., Recirculation Line Water Temperature – High) and administrative controls ensure that this flow path remains isolated to prevent unexpected loss of inventory via this flow path.

This Function isolates the Group 3 shutdown cooling valves.

ACTIONS

A Note has been provided to modify the ACTIONS related to primary containment 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,

(continued)

BASES

ACTIONS

C.1 (continued)

of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

D.1, D.2.1, and D.2.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time the associated MSLs may be isolated (Required Action D.1), and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. This Required Action will generally only be used if a Function 1.d channel is inoperable and untripped. The associated MSL(s) to be isolated are those whose Main Steam Line Flow-High Function channel(s) are inoperable. Alternately, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 2 within 8 hours. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

TECHNICAL CHANGES - MORE RESTRICTIVE

M.3 (cont'd) determined Operable (by performing a CHANNEL FUNCTIONAL TEST) within 1 hour after withdrawal of any control rod when RTP is ≤ 10%, not just when the withdrawal is for the purpose of making the reactor critical. This change is necessary to ensure the safety analysis assumptions concerning control rod worth are maintained by ensuring the RWM is Operable during any potential change in control rod worth. This is an additional restriction on plant operation.

M.4 With the RWM inoperable, the CTS 3.3.L Action allows control rod movement to continue provided a second licensed operator or other qualified member of the technical staff verifies control rod movement is in compliance with the prescribed control rod sequence. In ITS 3.3.2.1, with the RWM inoperable during a reactor startup, continued movement of control rods will only be allowed if ≥ 12 control rods are withdrawn (ITS 3.3.2.1 Required Action C.2.1.1) or if a startup with RWM inoperable has not been performed in the last 12 months (ITS 3.3.2.1 Required Action C.2.1.2). These new requirements are being added to ensure the RWM is reliable. These changes are additional restrictions on plant operation. D

M.5 A new RWM Surveillance has been added (proposed SR 3.3.2.1.6) to verify the automatic enabling point of the RWM. This SR ensures that the RWM is not inadvertently bypassed with power level ≤ 10% RTP. This is an additional restriction on plant operation to ensure proper operation of the RWM. B

M.6 A new RWM Surveillance has been added (proposed SR 3.3.2.1.9) to verify the bypassing and position of control rods required to be bypassed (taken out of service) in RWM by a second licensed operator or other qualified member of the technical staff. When a control rod is taken out of service in the RWM, if the control rod is fully inserted, the RWM provides an insert and withdraw block to the control rod. If the control rod is not fully inserted, the RWM provides only a withdraw block to the control rod. This is required prior to and during the movement of control rods bypassed in RWM. This is an additional restriction on plant operation to ensure proper operation of the RWM. C

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS Table 3.2.E-1 Note (a) states that the RBM shall be automatically bypassed when a peripheral control rod is selected. This system design detail is proposed to be relocated to the UFSAR. This design detail is not necessary to be included in the Technical Specifications to ensure the OPERABILITY of the RBM

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd) reactor is operating on a LIMITING CONTROL ROD PATTERN, have been deleted. The definition of LIMITING CONTROL ROD PATTERN is also being deleted. Since a LIMITING CONTROL ROD PATTERN is operation on a power distribution limit (such as APLHGR or MCPR), the condition is extremely unlikely. The status of power distribution limits does not affect the Operability of the RBM and therefore, no additional requirements on the RBM System are required (e.g., that it be tripped within one hour with a channel inoperable while on a LIMITING CONTROL ROD PATTERN). Adequate requirements on power distribution limits are specified in the LCOs in Section 3.2. Furthermore, due to the improbability of operating exactly on a thermal limit, the CTS Action and Surveillance Requirement would almost never be required. In addition, since the Surveillance Requirement is not specific as to when "prior to," and could be satisfied by the initial Surveillance that detected the LIMITING CONTROL ROD PATTERN has been achieved, its deletion is not safety significant.

L.4 CTS 3/4.3.L Applicability requires OPERABILITY of the RWM in OPERATIONAL MODE(s) 1 and 2 when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER. It is proposed to reduce the Applicability for RWM OPERABILITY (proposed ITS Table 3.3.2.1-1 footnote (b)) from ≤ 20% RTP to ≤ 10% RTP. This change will also result in a corresponding reduction in the power level identified in CTS 4.3.L.3 (ITS SR 3.3.2.1.3) for demonstrating the RWM OPERABLE (see Discussion of Change L.2 above). In addition, the power level identified in proposed ITS SRs 3.3.2.1.2 and 3.3.2.1.6 has been selected consistent with the proposed RWM Applicability of ≤ 10% RTP (see Discussion of Changes M.3, M.5, and L.2 above). The RWM serves to enforce pre-stored control rod withdrawal sequences to minimize the control rod worths during reactor startups. The lower control rod worths result in lower fuel enthalpy values, which mitigate the consequences of a Control Rod Drop Accident (CRDA). The RWM also generates rod blocks if a deviation from a programmed sequence is detected. This change essentially reduces the power level at which the RWM must be OPERABLE to ensure that the initial conditions of the CRDA are not violated. The NRC has approved, in the Safety Evaluation Report for Amendment 17 to NEDE-24011-P-A, "Acceptance for Referencing the Licensing Topical Report NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," dated December 27, 1988, the use of a ≤ 10% RTP Applicability for the RWM subject to the existence of analyses which "demonstrate that no significant rod drop accident (RDA) can occur above 10 percent power." Siemens Power Corporation (SPC) has performed CRDA analyses for the SPC fuel in the Dresden 2 and 3 reactors in support of reducing the RWM Applicability to ≤ 10% RTP. The analyses results show that the

TECHNICAL CHANGES - LESS RESTRICTIVE

L.4 (cont'd) consequences of a CRDA above 10% RTP are mitigated by factors which reduce available rod worths and enhance the effective actions of the feedback mechanisms. The SPC CRDA analysis methodology was explicitly reviewed and approved by the NRC and, based on this methodology, SPC has concluded that the predicted consequences for the CRDA above zero power conditions would be reduced. As a result, SPC further concluded that the ≤ 10% RTP Applicability for the RWM is adequate for reactors containing SPC fuel and that the RWM is not needed above 10% RTP. Since the SPC analyses demonstrate that the consequences for a CRDA above zero power would be reduced, it follows that no significant CRDA would occur above 10% RTP and the NRC's approval criterion for use of a ≤ 10% RTP Applicability for the RWM is satisfied. Therefore, the proposed change reducing the Applicability for RWM OPERABILITY from < 20% RTP to ≤ 10% RTP is considered acceptable.

RELOCATED SPECIFICATIONS

R.1 The SRM, IRM, Scram Discharge Volume, and APRM control rod blocks of CTS 3/4.2.E function to prevent positive reactivity insertion under conditions approaching those where RPS actuation may be expected. However, no design basis accident or transient takes credit for rod block signals initiated by this instrumentation. Further, the evaluation summarized in NEDO-31466 determined the loss of this instrumentation to be a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for these Functions in CTS 3/4.2.E did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the Dresden 2 and 3 Technical Specifications and have been relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the Dresden 2 and 3 UFSAR at ITS implementation. Changes to the TRM will be controlled in accordance with 10 CFR 50.59.

<CTS>

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

<3.1.A>
<T3.1.A-1>
<2.2.A>
<T2.2.A-1>

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

<Appl 3.1.A>
<T3.1.A-1>
<T3.1.A-1 Footnote (d)>
<Appl 2.2.A>

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

----------NOTE---------- (5) 15

(1.) Separate Condition entry is allowed for each channel.

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC A.3> <3.1.A Act 1> <3.1.A Act 2> <3.1.A Act 2.c> <2.2.A Act>	A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 Place associated trip system in trip.	12 hours 12 hours
<DOC A.3> <3.1.A Act 2> <3.1.A Act 2.b> <2.2.A Act>	B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours
<DOC A.3> <3.1.A Act 2.a> <3.1.A Act 3> <2.2.A Act>	C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

15

2. When Functions 2.b and 2.c channels are inoperable due to APRM indication not within limits, entry into associated Conditions and Required Actions may be delayed for up to 2 hours if the APRM is indicating a lower power value than the calculated power, and for up to 12 hours if the APRM is indicating a higher power value than the calculated power. (D) (C)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
17 15 <T4.1.A-1> SR 3.3.1.1.~~11~~	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until ~~12~~ 24 hours after entering MODE 2. 2 3. For Function 2.b, not required for the flow portion of the channels 6 ---------- Perform CHANNEL CALIBRATION.	184 days
17 16 <T4.1.A-1> SR 3.3.1.1.~~12~~	Perform CHANNEL FUNCTIONAL TEST.	~~[18]~~ 24 months 1
17 17 <T4.1.A-1> <DOC L.8> <T4.1.A-1 Footnote (h)> SR 3.3.1.1.~~13~~	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Function ~~1~~ .a, not required to be performed when entering MODE 2 from MODE 1 until ~~12~~ 24 hours after entering MODE 2. 14 2 ---------- Perform CHANNEL CALIBRATION.	24 1 ~~[18]~~ months
~~SR 3.3.1.1.14~~	~~Verify the APRM Flow Biased Simulated Thermal Power—High time constant is ≤ [7] seconds.~~	~~[18] months~~ 6
6 18 <4.1.A.2> SR 3.3.1.1.~~15~~	Perform LOGIC SYSTEM FUNCTIONAL TEST.	~~[18]~~ 24 months 1

(continued)

<CTS>

< moved to Page 3.3-4 >

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
17 <DOC M.3> SR 3.3.1.1.~~16~~ 14	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ ~~30~~ 45% RTP.	~~18 months~~ 92 days 1
17 <4.1.A.3> <DOC M.1> SR 3.3.1.1.~~17~~ 19	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Function 5 "n" equals 4 channels for the purpose of determining ~~the~~ the STAGGERED TEST BASIS Frequency. ---------- 14	
	Verify the RPS RESPONSE TIME is within limits.	24 1 ~~18~~ months on a STAGGERED TEST BASIS

D

all changes are [1] unless otherwise identified

[16] TSTF-264 changes not adopted

<T3.1.A-1>
<T4.1.A-1>
<T2.2.A-1>
<DOC L.8>

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.12 SR 3.3.1.1.16	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.12 SR 3.3.1.1.15 SR 3.3.1.1.16	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	3	H	SR 3.3.1.1.15 SR 3.3.2.2.1	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 20 % RTP
b. Flow Biased Simulated Thermal Power – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 0.58 W + 62 % RTP and ≤ 115.5 % RTP(b)

[14] [9] Neutron Flux [3] add SR 3.3.1.1.5 [16] add SR 3.3.1.1.17

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.58 W + 62 % ≤ 0.58 W RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." and ≤ 118.5 % 59.2

all changes are [1] unless otherwise identified

<T3.1.A-1>
<T4.1.A-1>
<T2.2.A-1>

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux – High	1	[2]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17	≤ [120]% RTP
add SR 3.3.1.1.5					
d. Downscale	1	[2]	F	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.15	≥ [3]% RTP
e. Inop	1,2	[2]	G	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.15	NA
3. Reactor Vessel Steam Dome Pressure – High	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.10 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ [1055] psig
add SR 3.3.1.1.8					
add SR 3.3.1.1.13					
4. Reactor Vessel Water Level – Low [Level 3]	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	≥ [-161] inches
5. Main Steam Isolation Valve – Closure	1	[8]	F	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ [10]% closed
6. Drywell Pressure – High	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.15 SR 3.3.1.1.16	≤ [1.92] psig
add SR 3.3.1.1.13					
add SR 3.3.1.1.19					

(continued)

(c) With reactor pressure ≥ 600 psig.

<CTS>

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC M.4> <3.3.L Act>	C. (continued)	C.2.1.1 Verify ≥ 12 rods withdrawn.	Immediately
		OR	
		C.2.1.2 Verify by administrative methods that startup with RWM inoperable has not been performed in the last ~~calendar year~~. 12 months 7	Immediately
		AND	
		C.2.2 Verify movement of control rods is in compliance with ~~banked position withdrawal sequence (BPWS)~~ analyzed rod position sequence 1 by a second licensed operator or other qualified member of the technical staff.	During control rod movement
<3.3.L Act>	D. RWM inoperable during reactor shutdown.	D.1 Verify movement of control rods is in ~~accordance~~ compliance 2 with ~~BPWS~~ by a second licensed operator or other qualified member of the technical staff.	During control rod movement

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
<4.3.L.2>	SR 3.3.2.1.2	----------NOTE---------- Not required to be performed until 1 hour after any control rod is withdrawn at ≤ ~~[10]~~% RTP in MODE 2. ---------- Perform CHANNEL FUNCTIONAL TEST.	~~92~~ days 3
5 <4.3.L.3>	SR 3.3.2.1.3	----------NOTE---------- Not required to be performed until 1 hour after THERMAL POWER is ≤ ~~[10]~~% RTP in MODE 1. ---------- Perform CHANNEL FUNCTIONAL TEST.	~~92~~ days 3
Insert SR 3.3.2.1.4 from page 3.3-19			
5 <DOC M.2>	SR 3.3.2.1.~~4~~ 5	----------NOTE---------- Neutron detectors are excluded. ---------- Verify the RBM ~~[is not bypassed when THERMAL POWER is ≥ 30% RTP and when a peripheral control rod is not selected.]~~ ~~a. Low Power Range—Upscale Function is not bypassed when THERMAL POWER is ≥ 29% and ≤ 64% RTP. b. Intermediate Power Range—Upscale Function is not bypassed when THERMAL POWER is > 64% and ≤ 84% RTP. c. High Power Range—Upscale Function is not bypassed when THERMAL POWER is > 84% RTP.~~	92 days ~~[18] months~~ 3 D D
5 <DOC M.5>	SR 3.3.2.1.~~5~~ 6	Verify the RWM is not bypassed when THERMAL POWER is ≤ ~~[10]~~% RTP.	~~[18]~~ 24 months 3

(continued)

<CTS>

<T3.2.E-1>
<DOC M.1>
<T4.2.E-1>
<DOC M.2>
<3.3.L>
<Appl 3.3.L>
<3.3.M>
<Appl 3.3.M>
<DOC M.6>

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range – Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	≤ [115.5/125] divisions of full scale
b. Intermediate Power Range – Upscale	(b)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	≤ [109.7/125] divisions of full scale
c. High Power Range – Upscale	(c),(d)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	≤ [105.9/125] divisions of full scale
d. Inop	(d),(e)	2	SR 3.3.2.1.1	NA
e. Downscale	(d),(g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	≥ [93/125] divisions of full scale
f. Bypass Time Delay	(d),(e)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.7	≤ [2.0] seconds
2. Rod Worth Minimizer	(f),(f)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.8	NA
3. Reactor Mode Switch – Shutdown Position	(g)	2	SR 3.3.2.1.9	NA

(a) THERMAL POWER ≥ [29]% and ≤ [64]% RTP and MCPR < 1.70.

(b) THERMAL POWER > [64]% and ≤ [84]% RTP and MCPR < 1.70.

(c) THERMAL POWER > [84]% and < 90% RTP and MCPR < 1.70.

(d) THERMAL POWER ≥ 90% RTP and MCPR < 1.40.

(e) THERMAL POWER ≥ [64]% and < 90% RTP and MCPR < 1.70.

(f) With THERMAL POWER ≤ [10]% RTP.

(g) Reactor mode switch in the shutdown position.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.3.2.1 - CONTROL ROD BLOCK INSTRUMENTATION

1. The proper Dresden 2 and 3 plant specific nomenclature/value/design requirements have been provided.

2. Editorial change made to be consistent with Required Action C.2.2.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. ISTS SR 3.3.2.1.4 and ISTS Table 3.3.2.1-1, Note (a) have been modified and ISTS Table 3.3.2.1-1, Functions 1.b, 1.c, and 1.f, including Notes (b), (c), (d), and (e) have been deleted to be consistent with the Dresden 2 and 3 RBM design. The RBM design in the ISTS is based on a "Post-ARTS" RBM design. Dresden 2 and 3 has not installed the "ARTS" RBM modification. In addition, the requirements have been renumbered, where applicable, to reflect the deletions.

5. ISTS SR 3.3.2.1.7 has been renumbered as SR 3.3.2.1.4 and the bracketed Frequency has been changed from 18 months to 92 days consistent with the current licensing basis. The Surveillances have been reordered and renumbered as required.

6. A new Surveillance (ITS SR 3.3.2.1.9) has been added to ITS 3.3.2.1 consistent with current and proposed requirements in the LaSalle Unit 1 and 2 Technical Specifications. This change was added for consistency in the ComEd Boiling Water Reactor Technical Specifications.

7. Required Action C.2.1.2 has been modified to be consistent with the Bases. 1 (D)

〈T3.2.F-1〉
〈Doc M.1〉
〈T4.2.F-1〉

Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	
1. Reactor ~~Steam Dome~~ Vessel Pressure	2	E	2
2. Reactor Vessel Water Level	2	E	2
3. ~~Suppression Pool~~ Torus Water Level	2	E	6
4. Drywell Pressure	2	E	7
5. ~~Primary Containment Area~~ Drywell Monitors Radiation	2	F	2
6. Drywell Sump Level	2	F	7
7. Drywell Drain Sump Level	2	F	7
8. PCIV Position (Penetration Flow Path)	2 per penetration flow path[a][b]	E	2
9. Wide Range Neutron Flux	2	F	7
10. Drywell H_2 ~~& O_2~~ Analyzer (Concentration and Monitor)	2	E	2
11. ~~Containment H_2 & O_2~~ Drywell Analyzer	2	E	2
12. Primary Containment Pressure	2	F	7
13. ~~Suppression Pool~~ Torus Water Temperature	2	E	2
[Relief Valve Discharge Location]			8

(a) Not required for isolation valves whose associated penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) Monitoring each [relief valve discharge location]. 6

Reviewer Note: Table 3.3.3.1-1 shall be amended for each plant as necessary to list:

1. All Regulatory Guide 1.97, Type A instruments, and
2. All Regulatory Guide 1.97, Category 1, non-Type A instruments specified in the plant's Regulatory Guide 1.97, Safety Evaluation Report. 7

a. Wide Range	2	E	6
b. Narrow Range	2	E	2

a. Fuel Zone (Wide Range)	2	E	6
b. Medium Range	2	E	

TSTF-295

TSTF-295

<CTS>

1 1

3.3 INSTRUMENTATION

3.3.4 2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

<3.2.C>
<T3.2.C-1>
<T4.2.C-1>

LCO 3.3.4 2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level—Low Low ~~Level 2~~; and

b. Reactor Vessel Steam Dome Pressure—High.

2

<Appl 3.2.C> APPLICABILITY: MODE 1.

ACTIONS

NOTE
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.2.C Act 1> <3.2.C Act 2> <3.2.C Act 3> <Doc M.2> <3.2.C Act 4> A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status.	14 days
	OR	
	A.2 NOTE: Not applicable if inoperable channel is the result of an inoperable breaker.	
	Place channel in trip.	14 days

(continued)

<CTS>

Insert Functions 2.g, 2.h, 2.i, 2.j and 2.k

<T3.2.B-1>
<T4.2.B-1>
<DOC M.1>

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
a. Low Pressure Coolant Injection Pump Discharge Flow – Low (Bypass)	1,2,3, 4(a), 5(a)	~~[1]~~ ~~1~~ per ~~[loop]~~ loop	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ ~~[ ]~~ 1107 gpm ~~and~~ ~~≤ [ ] gpm~~
h. Manual Initiation	1,2,3, 4(a), 5(a)	[2] [1 per subsystem]	C	SR 3.3.5.1.6	NA
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level – Low Low, ~~Level 2~~	1, 2(a), 3(a)	~~[4]~~	B	SR 3.3.5.1.1 SR 3.3.5.1.2 ~~SR 3.3.5.1.3~~ SR 3.3.5.1.5 SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~	≥ ~~[-47]~~ -54.15 inches
b. Drywell Pressure – High	1, 2(a), 3(a)	~~[4]~~	B	~~SR 3.3.5.1.1~~ SR 3.3.5.1.2 ~~SR 3.3.5.1.3~~ ~~SR 3.3.5.1.4~~ SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~	≤ ~~[1.92]~~ 1.81 psig
c. Reactor Vessel Water Level – High, ~~Level 8~~	1, 2(a), 3(a)	~~[2]~~	C	SR 3.3.5.1.1 SR 3.3.5.1.2 ~~SR 3.3.5.1.3~~ SR 3.3.5.1.5 SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~	≤ ~~[56.3]~~ 46.2 inches
d. Condensate Storage Tank Level – Low (CCST)	1, 2(a), 3(a)	~~[2]~~ 2 per CCST	D	~~SR 3.3.5.1.1~~ SR 3.3.5.1.2 ~~SR 3.3.5.1.3~~ SR 3.3.5.1.6	≥ ~~[10]~~ 11.1158 ft for CCST 2/3 A and 7.5637 ft for CCST 2/3 B
e. Suppression Pool Water Level – High	1, 2(a), 3(a)	~~[2]~~	D	~~SR 3.3.5.1.1~~ SR 3.3.5.1.2 ~~SR 3.3.5.1.3~~ SR 3.3.5.1.5 SR 3.3.5.1.6	≤ ~~[186]~~ 15 ft 5.625 inches

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE. (ECCS) per LCO 3.5.2

(b) With reactor steam dome pressure > ~~150~~ psig.

all changes are [1] unless otherwise identified

SURVEILLANCE REQUIREMENTS

NOTES

1. Refer to Table 3.3.5.2-1 to determine which SRs apply for each RCIC Function. [3]

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 as follows: (a) for up to 6 hours for Functions 2 and 5; and (b) for up to 6 hours for Functions 1, 3, and 4 provided the associated Function maintains RCIC initiation capability.

Reactor Vessel Pressure – High

	SURVEILLANCE		FREQUENCY
	SR 3.3.5.2.1	Perform CHANNEL CHECK.	12 hours
<4.2.D.1> <T4.2.D-1>	SR 3.3.5.2.2 (1)	Perform CHANNEL FUNCTIONAL TEST.	[92] (31) days [2]
	SR 3.3.5.2.3	Calibrate the trip units.	[92] days
<4.2.D.1> <T3.2.D-1> <T4.2.D-1>	SR 3.3.5.2.4 (2)	Perform CHANNEL CALIBRATION.	92 days [2]
	SR 3.3.5.2.5 (3)	Perform CHANNEL CALIBRATION	[18] (24) months [2] B
<4.2.D.2>	SR 3.3.5.2.6 (4)	Perform LOGIC SYSTEM FUNCTIONAL TEST.	[18] (24) months [2] B

The Allowable Value shall be ≤ 1068 psig. D A B

NOTE
Not required for the time delay portion of the channel.

for the time delay portion of the channel, The Allowable Value shall be ≤ 17 seconds.

<CTS>

3.3 INSTRUMENTATION

Emergency Ventilation (CREV) 1

3.3.7.1 ~~Main~~ Control Room Environmental Control (MCREC) System Instrumentation

Two channels of the Reactor Building Ventilation System - High High Radiation Alarm Function 2

<DOC M.1> LCO 3.3.7.1 The [MCREC] System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

<DOC M.1> APPLICABILITY: According to Table 3.3.7.1-1.

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

ACTIONS

NOTE
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.7.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	B.1 Declare associated [MCREC] subsystem inoperable. AND B.2 Place channel in trip.	1 hour from discovery of loss of [MCREC] initiation capability in both trip systems 24 hours

2 2

(continued)

all changes are [2] unless otherwise identified

<CTS>

<T3.2.B-1>
<T4.2.B-1>
<T3.2.B-1 Footnote (g)>
<T3.2.B-1 Footnote (j)>

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER BUS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. ~~4.16 kV Emergency~~ Bus Undervoltage (Loss of Voltage)			
a. ~~Bus Undervoltage~~ 4160 V Essential Service System	~~[2]~~	~~[SR 3.3.8.1.1]~~ SR 3.3.8.1.2 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ ~~[2800]~~ 2796.85 V and ≤ ~~[ ]~~ 3063.20 V
b. ~~Time Delay~~	~~[2]~~	~~[SR 3.3.8.1.2]~~ ~~SR 3.3.8.1.3~~ ~~SR 3.3.8.1.4~~	~~≥ [ ] seconds and ≤ [6.5] seconds~~
2. ~~4.16 kV Emergency~~ Bus Undervoltage (Degraded Voltage)			
a. Bus Undervoltage / Time Delay	~~[2]~~	~~[SR 3.3.8.1.1]~~ SR 3.3.8.1.2 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ ~~[3850]~~ 3861 V and ≤ ~~[ ]~~ 3911 V
b. Time Delay (No LOCA)	~~[1]~~ 1	~~[SR 3.3.8.1.1]~~ SR 3.3.8.1.2 SR 3.3.8.1.5	≥ ~~[ ]~~ 2.9 seconds and ≤ ~~[6.5]~~ 3.21 seconds

with time delay ≥ 5.7 seconds and ≤ 8.3 seconds [5]

<CTS>

3.3 INSTRUMENTATION

3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

<3.9.6> LCO 3.3.8.2 Two RPS electric power monitoring assemblies shall be OPERABLE for each inservice RPS motor generator set or alternate power supply.

<Appl 3.9.6> APPLICABILITY: MODES 1, 2, and 3, MODES 4 and 5 (with any control rod withdrawn from a core cell containing one or more fuel assemblies). [1] [D]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.9.6 Act 1> A. One or both inservice power supplies with one electric power monitoring assembly inoperable.	A.1 Remove associated inservice power supply(s) from service.	72 hours
<3.9.6 Act 2> B. One or both inservice power supplies with both electric power monitoring assemblies inoperable.	B.1 Remove associated inservice power supply(s) from service.	1 hour
<DoC A.3> C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3.	12 hours
	AND	
	C.2 Be in MODE 4.	36 hours [1]

(continued)

All changes [1] unless noted otherwise

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~1. Rod Block Monitor (continued)~~

~~SL (Ref. 3). Therefore, under these conditions, the RBM is also not required to be OPERABLE.~~

2. Rod Worth Minimizer

The RWM enforces the ~~banked position withdrawal~~ analyzed rod position sequence (~~BPWS~~) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, ~~and 7~~ 8. The ~~BPWS~~ analyzed rod position sequence requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the ~~BPWS~~ analyzed rod position sequence are specified in LCO 3.1.6, "Rod Pattern Control."

The RWM Function satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36(c)(2)(ii).

Since the RWM is a ~~hardwired~~ system designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. ~~7~~ 9). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the ~~BPWS~~ analyzed rod position sequence. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the ~~BPWS~~ analyzed rod position sequence, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is ~~> 10%~~ ≤ 10% RTP. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel ~~damage~~ design limit during a CRDA (Refs. ~~5 and 11~~ 4, 9, 10 and 11). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical.

(continued)

BASES

ACTIONS

A.1 (continued)

reason, Required Action A.1 requires restoration of the inoperable channel to OPERABLE status. The Completion Time of 24 hours is based on the low probability of an event occurring coincident with a failure in the remaining OPERABLE channel.

B.1

If Required Action A.1 is not met and the associated Completion Time has expired, the inoperable channel must be placed in trip within 1 hour. If both RBM channels are inoperable, the RBM is not capable of performing its intended function; thus, one channel must also be placed in trip. This initiates a control rod withdrawal block, thereby ensuring that the RBM function is met.

The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

2

These requirements minimize the number of reactor startups initiated with the RWM inoperable.

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM was not performed in the last 12 months. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2. Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff.

4

during withdrawal of one or more of the first 12 control rods

D

task

1

2

(e.g., shift technical advisor or reactor engineer)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

~~Reviewer's Note: Certain Frequencies are based on approved topical reports. In order for a licensee to use these Frequencies, the licensee must justify the Frequencies as required by the staff SER for the topical report.~~ 5

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note to indicate that when an RBM 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 control rod block 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 analysis (Ref. 8) 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 a control rod block will be initiated when necessary. 4 12 1 D

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control ~~Multiplexing~~ "Relay Select Matrix" System input.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 92 days is based on reliability analyses (Ref. 8). 13 1 C D

TSTF-205

INSERT SR

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with C

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

the prescribed sequence and verifying a control rod block occurs. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and entry into MODE 1 when THERMAL POWER is ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

SR 3.3.2.1.4

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1, each within a specific power range. The power at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8. The 18 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass

(continued)

BASES

REFERENCES (continued)

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

9 ~~7~~. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

Reference 10 from previous page

13 ~~8~~. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

12 ~~9~~. GENE-770-06-1-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," ~~February 1991~~ December 1992.

11. EMF-2237 (P), "Dresden Units 2 and 3 Reduced Low Power Set Point Analysis for Control Rod Drop Accident," July 1999.

all changes are [1] unless otherwise identified

BASES

BACKGROUND (continued)

~~(one-out-of-two logic similar to the CST water level logic). To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.~~ [2]

~~The RCIC System provides makeup water to the reactor until the reactor vessel water level reaches the high water level (Level 8) trip (two-out-of-two logic), at which time the RCIC steam supply, steam supply bypass, and cooling water supply valves close (the injection valve also closes due to the closure of the steam supply valves). The RCIC System restarts if vessel level again drops to the low level initiation point (Level 2).~~

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The function of the RCIC System to provide ~~makeup coolant~~ core cooling to the reactor is used to respond to ~~transient~~ events. Although a main steam line isolation [2] The RCIC System is ~~not~~ an Engineered Safety Feature System, ~~and~~ no credit is taken in the ~~safety~~ accident analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the IC system, and therefore its instrumentation, ~~are included in the Technical Specifications as required by the NRC Policy Statement. Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.~~ satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). [3] [2] [4] [2] [D]

LCO

The OPERABILITY of the RCIC System instrumentation is dependent upon the OPERABILITY of the ~~individual instrumentation channel~~ Function~~s~~ ~~specified in Table 3.3.5.2-1~~. Each ~~Function~~ must have ~~a required number of OPERABLE channels with their~~ setpoint~~s~~ within the ~~specified~~ Allowable Value~~s, where appropriate~~. ~~A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.~~ The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

four channels of the Reactor Vessel Pressure - High

channel

its

specified in SR 3.3.5.2.2

Move to Page B 3.3-141 as indicated

~~Allowable Value~~s ~~are specified~~ for ~~each~~ RCIC System instrumentation Function, specified in the ~~Table~~ SR. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL

The

is

the

(continued)

All changes are [1] unless noted otherwise

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6.b. Reactor Vessel Water Level—Low, ~~Level 3~~ (continued) [4]

is bounded by breaks of the recirculation and MSL. The ~~RHR~~ Shutdown Cooling System isolation on ~~Level 3~~ (low RPV water level) supports actions to ensure that the RPV water level does not drop below the top of the active fuel during a vessel draindown event caused by a leak (e.g., pipe break or inadvertent valve opening) in the ~~RHR~~ Shutdown Cooling System.

Reactor Vessel Water Level—Low, ~~Level 3~~ signals are initiated from four ~~level~~ (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 (two channels per trip system) of the Reactor Vessel Water Level—Low, ~~Level 3~~ [4] Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. As noted (footnote ~~(c)~~ (b) [4] to Table 3.3.6.1-1), ~~only two channels of the Reactor Vessel Water Level—Low, Level 3 Function are required to be OPERABLE in MODES 4 and 5 (and must input into the same trip system), provided the RHR Shutdown Cooling System integrity is maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system.~~ (Insert 6.b) [A]

The Reactor Vessel Water Level—Low, ~~Level 3~~ Allowable Value was chosen to be the same as the RPS Reactor Vessel Water Level—Low, ~~Level 3~~ Allowable Value (LCO 3.3.1.1), since the capability to cool the fuel may be threatened.

The Reactor Vessel Water Level—Low, ~~Level 3~~ Function is only required to be OPERABLE in MODES 3, 4, and 5 to prevent this potential flow path from lowering the reactor vessel level to the top of the fuel. In MODES 1 and 2, another isolation (i.e., ~~Reactor Steam Dome Pressure~~ (Recirculation Line Water Temperature)—High) and administrative controls ensure that this flow path remains isolated to prevent unexpected loss of inventory via this flow path.

This Function isolates the Group ~~11~~ (3) valves. (shutdown cooling)

(continued)

BASES

ACTIONS

B.1 (continued)

risk while allowing time for restoration or tripping of channels.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

2

This Required Action will generally only be used if a Function 1.d channel is inoperable and untripped. The associated MSL(s) to be isolated are those whose Main Steam Line Flow - High Function channel(s) are inoperable. Alternately,

1D

D.1, D.2.1, and D.2.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). Alternately, the associated MSLs may be isolated (Required Action D.1), and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

2

E.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 2 within 8 hours.

8

4

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Verify source of unidentified LEAKAGE increase is not intergranular stress corrosion cracking susceptible material.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Leakage Detection Instrumentation

LCO 3.4.5 The following RCS leakage detection instrumentation shall be OPERABLE:

a. Drywell floor drain sump monitoring system; and

b. Primary containment atmospheric particulate sampling system.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Drywell floor drain sump monitoring system inoperable.	A.1	Restore drywell floor drain sump monitoring system to OPERABLE status.	24 hours
B. Primary containment atmospheric particulate sampling system inoperable.	B.1	Restore primary containment atmospheric particulate sampling system to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Perform primary containment atmospheric particulate sampling.	12 hours
SR 3.4.5.2	Perform a CHANNEL FUNCTIONAL TEST of drywell floor drain sump monitoring system instrumentation.	31 days
SR 3.4.5.3	Perform a CHANNEL CALIBRATION of drywell floor drain sump monitoring system instrumentation.	12 months

BASES (continued)

ACTIONS

A.1

With RCS unidentified or total LEAKAGE greater than the limits, actions must be taken to reduce the leak. Because the LEAKAGE limits are conservatively below the LEAKAGE that would constitute a critical crack size, 4 hours is allowed to reduce the LEAKAGE rates before the reactor must be shut down. If an unidentified LEAKAGE has been identified and quantified, it may be reclassified and considered as identified LEAKAGE; however, the total LEAKAGE limit would remain unchanged.

B.1 and B.2

An unidentified LEAKAGE increase of > 2 gpm within a 24 hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE increase within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE increase to within limits (i.e., reducing the LEAKAGE rate such that the current rate is less than the "2 gpm increase in the previous 24 hours" limit; either by isolating the source or other possible methods) is to verify the source of the unidentified leakage increase is not material susceptible to IGSCC. D

The 4 hour Completion Time is reasonable to properly reduce the LEAKAGE increase or verify the source before the reactor must be shut down without unduly jeopardizing plant safety. D

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met or if pressure boundary LEAKAGE exists, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable,

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

UFSAR, Section 3.1.2.4.1 (Ref. 1), requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2). Leakage detection systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected and also to supply quantitative measurement of leakage rates. The Bases for LCO 3.4.4, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action.

LEAKAGE from the RCPB inside the drywell is detected by at least one of the two monitored variables, such as measuring flow from the drywell floor drain sump and primary containment atmospheric particulate radioactivity level. Although alternate methods of detecting RCS LEAKAGE are available, the sole means of quantifying LEAKAGE in the drywell is the drywell floor drain sump monitoring system.

The drywell floor drain sump monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the Reactor Building Closed Cooling Water System, and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump. Leakage into the drywell floor drain sump is pumped through a piping header that penetrates the containment wall to the floor drain collector tank.

(continued)

BASES

BACKGROUND (continued)

Two drywell floor drain sump pumps take suction from the drywell floor drain sump and discharge to the Liquid Radioactive Waste Management Systems. The pumps alternate as lead and backup on each successive start. When a high level is reached in the floor drain sump, a level switch actuates to start the lead floor drain sump pump when the pump discharge valves are open. In the event the level continues to rise, a second level switch actuates to start the backup floor drain sump pump and initiates an alarm in the control room. When the level decreases to a low level, both floor drain sump pumps are stopped. A flow transmitter in the discharge line of the drywell floor drain sump pumps provides flow indication in the control room. In addition, a leak rate recorder is provided capable of identifying a 1 gpm change over an 8 hour period. The pumps can also be started from the control room.

The primary containment atmospheric particulate sampling system provides a means to monitor the primary containment atmosphere for airborne particulate radioactivity. An increase of radioactivity may be attributed to RCPB steam or reactor water LEAKAGE. The primary containment atmospheric particulate sampling system is not capable of quantifying LEAKAGE rates. The primary containment atmospheric particulate sampling system consists of a manifold rack that allows drywell atmospheric grab samples to be obtained for analysis and a continuous air monitor that contains particulate and charcoal filters for monitoring of the drywell atmosphere. D

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 3 and 4). The drywell floor drain sump monitoring system is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm of excess LEAKAGE in the control room. The primary containment atmospheric particulate sampling system provides a means to detect changes in LEAKAGE rates. D D

A control room alarm provided by the drywell floor drain sump monitoring system allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and D

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 5). Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

The drywell floor drain sump monitoring system satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii). The primary containment atmospheric particulate sampling system does not meet any criteria of 10 CFR 50.36(c)(2)(ii), since it is not installed instrumentation that indicates in the control room. However, it is maintained to be consistent with NUREG-1433.

LCO

The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, the flow monitoring portion of the system must be OPERABLE. The primary containment atmospheric particulate sampling system is available to the operators so closer examination can be made to determine the extent of any corrective action that may be required. Only one sampling method (either the manifold rack or the continuous air monitor) is required to meet the OPERABILITY requirements. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.

APPLICABILITY

In MODES 1, 2, and 3, the leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, other monitoring systems are normally available that will provide indication of changes in leakage.

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.4.1), operation may continue for 24 hours. The 24 hour Completion Time of Required Action A.1 is acceptable, based on operating

(continued)

BASES

ACTIONS

A.1 (continued)

experience, considering the alternative form of leakage detection that is normally available and the fact that the LEAKAGE is still being determined every 12 hours.

B.1

With the primary containment atmospheric particulate sampling system inoperable, operation may continue for 24 hours. The 24 hour Completion Time of Required Action B.1 is acceptable, based on operating experience, considering the alternative form of leakage detection that is normally available and the fact that the LEAKAGE is still being determined every 12 hours (SR 3.4.4.1).

C.1 and C.2

If the Required Action and associated Completion Time of Condition A or B cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires performance of a primary containment atmospheric particulate sample every 12 hours. This is performed by either removing and analyzing the particulate and charcoal filters from the continuous air monitor or by analyzing a grab sample.

SR 3.4.5.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the drywell floor drain sump monitoring system instrumentation. The test ensures that the system can perform its function in the desired manner. The test also verifies the relative accuracy of the instrument string. A

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.2

successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.5.3

This SR is for the performance of a CHANNEL CALIBRATION of the drywell floor drain sump monitoring system instrumentation channel (i.e., drywell floor drain sump pump discharge flow integrator). The calibration verifies the accuracy of the instrument string. The Frequency of SR 3.4.5.3 is based on the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Section 3.1.2.4.1.

2. Regulatory Guide 1.45, May 1973.

3. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.

4. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.

5. UFSAR, Section 5.2.5.6.4.

BASES

APPLICABILITY (continued)

shutdown cooling piping. Decay heat removal at reactor temperatures greater than or equal to the SDC cut-in permissive temperature is typically accomplished by condensing the steam in the main condenser.

The requirements for decay heat removal in MODE 3 below the cut-in permissive temperature and in MODE 5 are discussed in LCO 3.4.7, "Shutdown Cooling (SDC) System—Hot Shutdown"; LCO 3.9.8, "Shutdown Cooling (SDC)—High Water Level"; and LCO 3.9.9, "Shutdown Cooling (SDC)—Low Water Level."

ACTIONS

A Note has been provided to modify the ACTIONS related to SDC subsystems. Section 1.3, Completion Times, specifies 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 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 shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable SDC subsystem.

A.1

With one of the two required SDC subsystems inoperable, except as permitted by LCO Note 3, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore, an alternate method of decay heat removal must be provided. With both required SDC subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial SDC subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the

(continued)

BASES

ACTIONS

A.1 (continued)

functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will provide assurance of continued heat removal capability.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) the Condensate/Feed and Main Steam System and the Reactor Water Cleanup System (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System).

B.1 and B.2

With no required SDC subsystem and no recirculation pump in operation, except as permitted by LCO Notes 1 and 2, and until SDC or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the coolant circulation function and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability. C

During the period when the reactor coolant is being circulated by an alternate method (other than by the required SDC System or recirculation pump), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.

(continued)

A.1

3.6 - LIMITING CONDITIONS FOR OPERATION

G. Leakage Detection Systems

LCO 3.4.5 The following reactor coolant system leakage detection systems shall be OPERABLE:

1. The primary containment atmosphere particulate radioactivity sampling system, and

2. The drywell floor drain sump system.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

ACTION B) 1. With the primary containment atmosphere particulate radioactivity sampling system inoperable, restore the inoperable leak detection radioactivity sampling system to OPERABLE status within 24 hours; otherwise, be in HOT

ACTION C) SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION A) 2. With the drywell floor drain sump system inoperable, restore the drywell floor drain sump system to OPERABLE status within 24 hours; otherwise, be

ACTION C) in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

G. Leakage Detection Systems

The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

1. ~~Performing the leakage determinations of Specification 4.6.H.~~ A.2 add proposed SR 3.4.5.2 M.1 D

2. Performing a CHANNEL CALIBRATION

SR 3.4.5.3 of the drywell floor drain sump pump discharge flow integrator at least once per 18 months. LA.1 D

12 M.2

monitoring system A.3

D

D

A.1

3.6 - LIMITING CONDITIONS FOR OPERATION

H. Operational Leakage

Reactor coolant system leakage shall be limited to:

1. No PRESSURE BOUNDARY LEAKAGE.
2. ≤25 gpm total leakage averaged over any 24 hour surveillance period.
3. ≤5 gpm UNIDENTIFIED LEAKAGE.
4. ≤2 gpm increase in UNIDENTIFIED LEAKAGE within any period of 24 hours or less (Applicable in OPERATIONAL MODE 1 only).

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
2. With the reactor coolant system UNIDENTIFIED LEAKAGE or total leakage rate(s) greater than the above limit(s), reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

H. Operational Leakage

The reactor coolant system leakage shall be demonstrated to be within each of the limits by:

SR 3.4.5.1

1. Sampling the primary containment atmospheric particulate radioactivity at least once per 12 hours[a], and
2. Determining the primary containment sump flow rate at least once per 8 hours, not to exceed 12 hours.

See ITS 3.4.4

D

a Not a means of quantifying leakage.

LA.1 D

DISCUSSION OF CHANGES
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The requirement in CTS 4.6.G.1 to perform the leakage determinations of CTS 4.6.H has been deleted since it duplicates the requirement of CTS 4.6.H.2 (proposed ITS SR 3.4.4.1). Therefore, this change is considered administrative.

A.3 The Dresden 2 and 3 design includes a single qualified leakage detection system, although other methods of RCS leakage detection are available. The words, "drywell floor drain sump pump discharge flow integrator," in CTS 4.6.G.2 are proposed to be replaced with the qualified detection system name, "drywell floor drain sump monitoring system," for clarification and to provide consistency with the proposed changes to the LCO and ACTIONS. Therefore, the words, "monitoring system," have been added to CTS 4.6.G.2. Since this change only provides plant specific clarification of the existing requirements, the change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 ITS SR 3.4.5.2 has been added to CTS 4.6.G to require a CHANNEL FUNCTIONAL TEST to be performed on the drywell floor drain sump monitoring system on a 31 day frequency. This requirement ensures the monitor can perform its function and verifies the relative accuracy of the instrument string. This is an added requirement necessary to help ensure the RCS leakage detection instrumentation is maintained OPERABLE and therefore is considered more restrictive. D

M.2 The Frequency of the CHANNEL CALIBRATION requirement for CTS 4.6.G.2, Drywell Floor Drain Sump Monitoring System, has been increased from 18 months to 12 months (proposed ITS SR 3.4.5.3). The proposed Frequency is acceptable since it is consistent with current plant calculations. This change to the CTS requirement constitutes a more restrictive change to help ensure that the drywell floor drain sump monitoring system is maintained OPERABLE. D

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The detail in CTS 4.6.G.2 of what Drywell Floor Drain Sump Monitoring System instrumentation (pump discharge flow integrator) is subject to a CHANNEL CALIBRATION and the detail in CTS 4.6.H.1 footnote (a) that the primary containment atmospheric particulate sampling system is not a means for quantifying leakage is proposed to be relocated to the Bases. These details are not necessary to ensure that a CHANNEL CALIBRATION is performed or that the primary containment atmospheric particulate sample is taken. Proposed SR 3.4.5.3, in conjunction with the Bases, requires the CHANNEL CALIBRATION to verify the accuracy of the drywell floor drain sump pump discharge flow integrator instrument string. This is consistent with the intent of CTS 4.6.G.2 and provides assurance that the instrumentation is OPERABLE when required. Proposed SR 3.4.5.1 ensures a sample is taken and LCO 3.4.5.b ensures the primary containment atmospheric particulate sampling system is OPERABLE. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

None

RELOCATED SPECIFICATIONS

None

<CTS>

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.H Act 3.a> [3]	B. (continued)	B.2 Verify source of unidentified LEAKAGE increase is not ~~service sensitive type 304 or type 316 austenitic stainless steel~~ intergranular stress corrosion cracking.	4 hours
<3.6.H Act 1> <3.6.H Act 2> <3.6.H Act 3.b>	C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.6.H>	SR 3.4.4.1	Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	~~8~~ 12 hours [4]

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Leakage Detection Instrumentation

<3.6.G> LCO 3.4.6 The following RCS leakage detection instrumentation shall be OPERABLE:

a. Drywell floor drain sump monitoring system; ~~[and]~~

b. ~~One channel of either~~ primary containment atmospheric particulate ~~or atmospheric gaseous monitoring~~ sampling system~~s~~ ~~[and~~

c. ~~Primary containment air cooler condensate flow rate monitoring system].~~

<Appl 3.6.G> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.G Act 2> A. Drywell floor drain sump monitoring system inoperable.	~~----------NOTE----------~~ ~~LCO 3.0.4 is not applicable.~~ ~~----------~~	TSTF-60 not adopted
	A.1 Restore drywell floor drain sump monitoring system to OPERABLE status.	~~30 days~~ 24 hours

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.6.6 Act 1 B. ~~Required~~ primary containment atmospheric ~~monitoring~~ system inoperable. Particulate sampling	~~----------NOTE---------- LCO 3.0.4 is not applicable. ----------~~ ~~B.1 Analyze grab samples of primary containment atmosphere.~~ ~~AND~~ B.2 Restore ~~required~~ primary containment atmospheric ~~monitoring~~ system to OPERABLE status.	~~Once per 12 hours~~ 24 hours ~~30 days~~
~~C. Primary containment air cooler condensate flow rate monitoring system inoperable.~~	~~C.1 ---------NOTE-------- Not applicable when required primary containment atmospheric monitoring system is inoperable. ---------- Perform SR 3.4.6.1.~~	~~Once per 8 hours~~

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required primary containment atmospheric monitoring system inoperable. AND Primary containment air cooler condensate flow rate monitoring system inoperable.	----------NOTE---------- LCO 3.0.4 is not applicable. ---------- D.1 Restore required primary containment atmospheric monitoring system to OPERABLE status. OR D.2 Restore primary containment air cooler condensate flow rate monitoring system to OPERABLE status.	30 days 30 days
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours
F. All required leakage detection systems inoperable.	F.1 Enter LCO 3.0.3.	Immediately

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
1	SR 3.4.~~6~~ 5.1	Perform a ~~CHANNEL CHECK of required~~ primary containment atmospheric ~~monitoring system~~ particulate sampling.	12 hours 2	D
1 <Doc M.1>	SR 3.4.~~6~~ 5.2	Perform a CHANNEL FUNCTIONAL TEST of ~~required leakage detection~~ drywell floor drain sump monitoring system instrumentation. 5	31 days	D
1 <4.6.G>	SR 3.4.~~6~~ 5.3	Perform a CHANNEL CALIBRATION of ~~required leakage/detection~~ drywall floor drain sump monitoring system instrumentation. 5	~~[18]~~ 12 months 6	D

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

1. BWR ISTS, NUREG-1433, Revision 1, Specification 3.4.5, "Reactor Coolant System Pressure Isolation Valve (PIV) Leakage," has not been incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

2. Changes have been made to reflect plant specific nomenclature and current licensing basis requirements.

3. The bracketed requirement/information has been deleted since it is not applicable to Dresden 2 and 3. The following requirements have been renumbered, where applicable, to reflect this deletion.

4. Not used. D

5. The Dresden 2 and 3 design includes a single qualified leakage detection system, although other methods of RCS leakage detection are available. The words, "required leakage detection," in ITS SRs 3.4.5.2 and 3.4.5.3 have been replaced with the qualified detection system name, "drywell floor drain sump monitoring system" for clarification and to provide consistency with the proposed changes to the LCO and ACTIONS. D

6. The brackets have been removed and the proper plant specific information/value has been provided.

BASES (continued)

ACTIONS

A.1

With RCS unidentified or total LEAKAGE greater than the limits, actions must be taken to reduce the leak. Because the LEAKAGE limits are conservatively below the LEAKAGE that would constitute a critical crack size, 4 hours is allowed to reduce the LEAKAGE rates before the reactor must be shut down. If an unidentified LEAKAGE has been identified and quantified, it may be reclassified and considered as identified LEAKAGE; however, the total LEAKAGE limit would remain unchanged.

B.1 and B.2

An unidentified LEAKAGE increase of > 2 gpm within a 4 hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE increase within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE increase to within limits (i.e., reducing the LEAKAGE rate such that the current rate is less than the "2 gpm increase in the previous 4 hours" limit; either by isolating the source or other possible methods) is to evaluate service sensitive type 304 and type 316 austenitic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids and determine it is not the source of the increased LEAKAGE. This type piping is very susceptible to IGSCC.

verify the source of the unidentified leakage increase is not material

The 4 hour Completion Time is reasonable to properly reduce the LEAKAGE increase or verify the source before the reactor must be shut down without unduly jeopardizing plant safety.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met or if pressure boundary LEAKAGE exists, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable,

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

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

Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2). Leakage detection systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected and also to supply quantitative measurement of leakage rates. The Bases for LCO 3.4.4, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action.

LEAKAGE from the RCPB inside the drywell is detected by at least one of two or three independently monitored variables, such as sump level changes and drywell gaseous and particulate radioactivity levels. The primary means of quantifying LEAKAGE in the drywell is the drywell floor drain sump monitoring system.

The drywell floor drain sump monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the Closed Cooling Water System, and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump. The primary containment floor drain sump has transmitters that supply level indications in the main control room.

The floor drain sump level indicators have switches that start and stop the sump pumps when required. A timer starts each time the sump is pumped down to the low level setpoint.

(continued)

BASES

BACKGROUND (continued)

If the sump fills to the high level setpoint before the timer ends, an alarm sounds in the control room, indicating a LEAKAGE rate into the sump in excess of a preset limit.

Insert BKGD 1

A flow indicator in the discharge line of the drywell floor drain sump pumps provides flow indication in the control room. The pumps can also be started from the control room.

Insert BKGD 2

The primary containment air monitoring system continuously monitor the primary containment atmosphere for airborne particulate and gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The primary containment atmosphere particulate and gaseous radioactivity monitoring systems are not capable of quantifying LEAKAGE rates, but are sensitive enough to indicate increased LEAKAGE rates of 1 gpm within 1 hour. Larger changes in LEAKAGE rates are detected in proportionally shorter times (Ref. 3).

Insert BKD 3

Condensate from four of the six primary containment coolers is routed to the primary containment floor drain sump and is monitored by a flow transmitter that provides indication and alarms in the control room. This primary containment air cooler condensate flow rate monitoring system serves as an added indicator, but not quantifier, of RCS unidentified LEAKAGE.

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. A and B). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm of excess LEAKAGE in the control room.

A control room alarm allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. B). Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

The primary containment atmospheric particulate sampling system provides a means to detect changes in LEAKAGE rates.

(continued)

Z INSERT BKGD 1

Two drywell floor drain sump pumps take suction from the drywell floor drain sump and discharge to the Liquid Radioactive Waste Management Systems. The pumps alternate as lead and backup on each successive start. When a high level is reached in the floor drain sump, a level switch actuates to start the lead floor drain sump pump when the pump discharge valves are open. In the event the level continues to rise, a second level switch actuates to start the backup floor drain sump pump and initiates an alarm in the control room. When the level decreases to a low level, both floor drain sump pumps are stopped.

Z INSERT BKGD 2

In addition, a leak rate recorder is provided capable of identifying a 1 gpm change over an 8 hour period.

Z INSERT BKGD 3

The primary containment atmospheric particulate sampling system consists of a manifold rack that allows drywell atmospheric grab samples to be obtained for analysis and a continuous air monitor that contains particulate and charcoal filters for monitoring of the drywell atmosphere. D

The primary containment atmospheric particulate sampling system does not meet any criteria of 10 CFR 50.36 (c)(2)(ii), since it is not installed instrumentation that indicates in the control room. However, it is maintained to be consistent with NUREG-1433.

BASES

The drywell floor drain sump monitoring system

APPLICABLE SAFETY ANALYSES (continued)

RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement. 10 CFR 50.36 (c)(2)(ii)

primary containment atmospheric particulate sampling system is available

LCO

The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, either the flow monitoring or the sump level monitoring portion of the system must be OPERABLE. The other monitoring systems provide early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.

can

Only one sampling method (either the containment rack or the continuous air monitor) is required to meet the OPERABILITY requirements.

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the primary containment atmospheric activity monitor (and the primary containment air cooler condensate flow rate monitor) will provide indication of changes in leakage.

other monitoring systems are normally available that.

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 8 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

12 — 24 hours — 24 hour — alternative — is normally — and the fact that the LEAKAGE is still being determined every 12 hours

(continued)

BASES

ACTIONS (continued)

B.1 ~~and B.2~~

INSERT B.1

~~With both gaseous and particulate primary containment atmospheric monitoring channels inoperable, grab samples of the primary containment atmosphere must be taken and analyzed to provide periodic leakage information. [Provided a sample is obtained and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of at least one of the required monitors.] [Provided a sample is obtained and analyzed every 12 hours, the plant may continue operation since at least one other form of drywell leakage detection (i.e., air cooler condensate flow rate monitor) is available.]~~

~~The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. The 30 day Completion Time for restoration recognizes that at least one other form of leakage detection is available.~~

~~The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate primary containment atmospheric monitoring channels are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.~~

~~C.1~~

~~With the required primary containment air cooler condensate flow rate monitoring system inoperable, SR 3.4.6.1 must be performed every 8 hours to provide periodic information of activity in the primary containment at a more frequent interval than the routine Frequency of SR 3.4.7.1. The 8 hour interval provides periodic information that is adequate to detect LEAKAGE and recognizes that other forms of leakage detection are available. However, this Required Action is modified by a Note that allows this action to be not applicable if the required primary containment atmospheric monitoring system is inoperable. Consistent with SR 3.0.1, Surveillances are not required to be performed on inoperable equipment.~~

(continued)

2 INSERT B.1

With the primary containment atmospheric particulate sampling system inoperable, operation may continue for 24 hours. The 24 hour Completion Time of Required Action B.1 is acceptable, based on operating experience, considering the alternative form of leakage detection that is normally available and the fact that the LEAKAGE is still being determined every 12 hours (SR 3.4.4.1).

D

BASES

ACTIONS (continued)

D.1 and D.2

With both the primary containment gaseous and particulate atmospheric monitor channels and the primary containment air cooler condensate flow rate monitor inoperable, the only means of detecting LEAKAGE is the drywell floor drain sump monitor. This condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate primary containment atmospheric monitoring channels and air cooler condensate flow rate are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

E.1 and E.2

If any Required Action of Condition A, B, C, or D cannot be met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

F.1

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

This is performed either by removing and analyzing the particulate and charcoal filters from the continuous air monitor or by analyzing a grab sample.

SR 3.4.6.1

This SR is for the performance of a CHANNEL CHECK of the required primary containment atmospheric monitoring system. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.6.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.6.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section [5.2.7.2.1].
4. GEAP-5620, April 1968.
5. NUREG-75/067, October 1975.
6. FSAR, Section [5.2.7.5.2].

all changes are [1] unless otherwise indicated

BASES

APPLICABILITY (continued)

the steam in the main condenser. ~~Additionally, in MODE 4 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the RHR shutdown cooling subsystem into operation.~~ [3]

The requirements for decay heat removal in MODE 3 below the cut-in permissive ~~pressure~~ temperature and in MODE 5 are discussed in LCO 3.4.8, "~~Residual Heat Removal (RHR)~~ SDC Shutdown Cooling System—Hot Shutdown"; LCO 3.9.8, "~~Residual Heat Removal (RHR)~~ SDC—High Water Level"; and LCO 3.9.9, "~~Residual Heat Removal (RHR)~~ Shutdown Cooling (SDC)—Low Water Level." [2] [6]

ACTIONS

A Note has been provided to modify the ACTIONS related to ~~RHR shutdown cooling~~ SDC subsystems. Section 1.3, Completion Times, specifies 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 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 shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable ~~RHR shutdown cooling~~ SDC subsystem.

A.1

With one of the two required ~~RHR shutdown cooling~~ SDC subsystems inoperable, except as permitted by LCO Note 2, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore, an alternate method of decay heat removal must be provided. With both ~~RHR shutdown cooling~~ SDC subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial ~~RHR shutdown cooling~~ SDC subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour [3] [D]

required

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify, for each required ECCS injection/spray subsystem, the: a. Suppression pool water level is ≥ 10 ft 4 inches; or b. ----------NOTE---------- Only one required ECCS injection/spray subsystem may take credit for this option during OPDRVs. ---------- Contaminated condensate storage tanks water volume is ≥ 140,000 available gallons.	12 hours
SR 3.5.2.2	Verify, for each required ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.2.3	Verify each required ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ISOLATION CONDENSER (IC) SYSTEM

B 3.5.2 ECCS – Shutdown

BASES

BACKGROUND	A description of the Core Spray (CS) System and the Low Pressure Coolant Injection (LPCI) System is provided in the Bases for LCO 3.5.1, "ECCS – Operating."
APPLICABLE SAFETY ANALYSES	The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated loss of coolant accident (LOCA). The long term cooling analysis following a design basis LOCA (Ref. 1) demonstrates that only one low pressure ECCS injection/spray subsystem is required, post LOCA, to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. It is reasonable to assume, based on engineering judgement, that while in MODES 4 and 5, one low pressure ECCS injection/spray subsystem can maintain adequate reactor vessel water level. To provide redundancy, a minimum of two low pressure ECCS injection/spray subsystems are required to be OPERABLE in MODES 4 and 5. The low pressure ECCS subsystems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two low pressure ECCS injection/spray subsystems are required to be OPERABLE. The low pressure ECCS injection/spray subsystems consist of two CS subsystems and two LPCI subsystems. Each CS subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or contaminated condensate storage tanks (CCSTs) to the reactor pressure vessel (RPV). Each LPCI subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or the CCSTs to the RPV. A single LPCI pump is required per subsystem because of the similar injection capacity in relation to a CS subsystem. In addition, in MODES 4 and 5, the LPCI System cross-tie valves are not required to be open.

(continued)

BASES

ACTIONS

C.1, C.2, D.1, D.2, and D.3 (continued)

secondary containment is indicated). OPERABILITY may be verified by an administrative check, or by examining logs or other information, to determine whether the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the Surveillance may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

The minimum water level of 10 ft 4 inches above the bottom of the suppression chamber required for the suppression pool is periodically verified to ensure that the suppression pool will provide adequate net positive suction head (NPSH) for the CS System and LPCI subsystem pumps, recirculation volume, and vortex prevention. With the suppression pool water level less than the required limit, all ECCS injection/spray subsystems are inoperable unless they are aligned to OPERABLE CCSTs.

When suppression pool level is < 10 ft 4 inches, the CS and LPCI subsystems are considered OPERABLE only if they can take suction from the CCSTs, and the CCSTs water volume is sufficient to provide the required NPSH and vortex prevention for the CS pump and LPCI pump. Therefore, a verification that either the suppression pool water level is ≥ 10 ft 4 inches or that required low pressure ECCS injection/spray subsystems are aligned to take suction from the CCSTs and the CCSTs contain ≥ 140,000 available gallons of water, equivalent to 23 ft in both CCSTs with the CCSTs crosstied, ensures that the required low pressure ECCS injection/spray subsystems can supply at least 140,000 gallons of makeup water to the RPV. The CS and LPCI suctions are uncovered at the 90,000 gallon level. However, as noted, only one required low pressure ECCS injection/spray subsystem may take credit for the CCST option during OPDRVs. During OPDRVs, the volume in the CCSTs may not provide adequate makeup if the RPV were

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 (continued)

completely drained. Therefore, only one low pressure ECCS injection/spray subsystem is allowed to use the CCSTs. This ensures the other required ECCS subsystem has adequate makeup volume.

The 12 hour Frequency of these SRs was developed considering operating experience related to suppression pool water level and CCST water level variations and instrument drift during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool or CCST water level condition.

SR 3.5.2.2, SR 3.5.2.4, and SR 3.5.2.5

The Bases provided for SR 3.5.1.1, SR 3.5.1.5, and SR 3.5.1.8 are applicable to SR 3.5.2.2, SR 3.5.2.4, and SR 3.5.2.5, respectively.

SR 3.5.2.3

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is appropriate because the valves are operated under procedural control and the probability of their being mispositioned during this time period is low.

REFERENCES

1. UFSAR, Section 6.3.3.4.1.

A.1

ITS 3.5.2

SR 3.5.2.2
SR 3.5.2.3
SR 3.5.2.4

3.5 - LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling System - Shutdown

LCO 3.5.2

low pressure ECCS injection/spray

At least two of the following four subsystems/loops shall be OPERABLE:

LA.1

1. One or both core spray (CS) subsystems with:

   a. An OPERABLE flow path capable of taking suction from at least one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

      1) From the suppression chamber, or

      2) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least 140,000 available gallons of water.

SR 3.5.2.1.b

2. One or both low pressure coolant injection (LPCI) subsystem loops with a subsystem loop comprised of:

   a. At least one OPERABLE LPCI pump, and

   b. An OPERABLE flow path capable of taking suction from at least one of the following water sources and transferring the water to the reactor vessel:

      1) From the suppression chamber, or

LA.1

4.5 - SURVEILLANCE REQUIREMENTS

B. Emergency Core Cooling System - Shutdown

SR 3.5.2.5

LD.1

The required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.A, except:

A.2

1. The LPCI subsystems cross-tie valves may be closed.

LA.1

2. Each LPCI pump develops the required flow when tested pursuant to Specification 4.0.E.

SR 3.5.2.4

M.1

add proposed flowrate and head conditions for one pump

D

A.1

ITS 3.5.2

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

(2) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least 140,000 available gallons of water. — A.1

SR 3.5.2.1.b

D

APPLICABILITY:

OPERATIONAL MODE(s) 4 and 5[a].

ACTION:

ACTION A — 1. With one of the above required subsystems/loops inoperable, restore at least two subsystems/loops to OPERABLE status within 4 hours or

ACTION B — suspend all operations with a potential for draining the reactor vessel.

ACTION C — 2. With both of the above required subsystems/loops inoperable, suspend CORE ALTERATIONS and all operations with a potential for draining the reactor vessel. Restore at least one subsystem/loop to OPERABLE status within 4 hours or establish — L.1

ACTION D — SECONDARY CONTAINMENT INTEGRITY within the next 8 hours — A.4 A.3

Applicability — a The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.10.G and 3.10.H. — A.5

A.1 ITS 3.5.2

3.5 - LIMITING CONDITIONS FOR OPERATION

C. Suppression Chamber

moved to ITS 3.6.2.2

The suppression chamber shall be OPERABLE: A.6

1. In OPERATIONAL MODE(s) 1, 2, and 3 with a contained water volume equivalent to a water level of ≥14' 6.5" above the bottom of the suppression chamber.

2. In OPERATIONAL MODE(s) 4 and 5[a] with a contained volume equivalent to a water level of ≥8' above the bottom of the suppression chamber, except that the suppression chamber level may be less than the limit provided that:

SR 3.5.2.1.a (10 ft, 4 inches) A.8 LA.1

a. No operations are performed that have a potential for draining the reactor vessel, L.2

b. The reactor mode switch is locked in the Shutdown or Refuel position, L.3

c. The condensate storage tank contains ≥140,000 available gallons of water, and

SR 3.5.2.1.b

Add Note to SR 3.5.2.1.b L.2

d. The ECCS systems are OPERABLE per Specification 3.5.B.

LCO 3.5.2

APPLICABILITY:

A.6 moved to ITS 3.6.2.2

OPERATIONAL MODE(s) 1, 2, 3, 4 and 5[a].

4.5 - SURVEILLANCE REQUIREMENTS

C. Suppression Chamber

The suppression chamber shall be determined OPERABLE by verifying:

1. For OPERATIONAL MODE(s) 1, 2 and 3, at least once per 24 hours, the water level to be ≥14' 6.5".

2. For OPERATIONAL MODE(s) 4 or 5[a], at least once per 12 hours: A.7 A.8

SR 3.5.2.1.a

a. The water level to be ≥8', or (10 ft, 4 inches)

SR 3.5.2.1.b

b. Verify the alternate conditions of Specification 3.5.C.2, or the conditions of footnote (a), to be satisfied. L.4 A.7

---

a The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specification 3.10.G and 3.10.H. A.5 M.2

Applicability

TECHNICAL CHANGES - LESS RESTRICTIVE

L.4 (cont'd) volume of water (the 12 hour Frequency will be retained as indicated in proposed SR 3.5.2.1.b), and the ECCS are OPERABLE per Specification 3.5.B. In the ITS, the requirements of 3/4.5.C and 3/4.5.B are incorporated in one Specification (ITS 3.5.2) and only the normal Surveillance Frequencies are proposed. This change is based on the fact that it is overly conservative to assume that systems or components are inoperable when a surveillance has not been performed. The opposite is in fact the case, the vast majority of surveillances demonstrate that systems or components in fact are operable. Therefore, even with low suppression pool level, the normal frequencies (e.g., LPCI testing in accordance with the Inservice Testing Frequency) are considered sufficient to ensure the OPERABILITY of the systems and that the parameters are within limits.

RELOCATED SPECIFICATIONS

None

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
<3.5.B.1.a.2)> <3.5.B.2.b.2)> <3.5.C.2> <3.5.C.2.c> <4.5.C.2>	SR 3.5.2.~~2~~ 3 1	Verify, for each required ~~core spray (CS)~~ ECCS injection/spray subsystem, the:	12 hours 3
	2 10 4	a. Suppression pool water level is ≥ ~~12~~ ft ~~2~~ inches; or	
	3 EC injection/spray	b. ----------NOTE---------- Only one required ~~CS~~ subsystem may take credit for this option during OPDRVs. ----------	
	1 Contaminated	5 1 volume Condensate storage tank water ~~level~~ is ≥ ~~12 ft~~ 140,000 available gallons 2	
<4.5.B>	SR 3.5.2.~~3~~ 3 2	Verify, for each required ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
<4.5.B>	SR 3.5.2.~~4~~ 3 3	~~----------NOTE---------- One LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned and not otherwise inoperable. ----------~~ 4	
		Verify each required ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

all changes are [2] unless otherwise identified

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM (ISOLATION CONDENSER (IC)) [1]

B 3.5.2 ECCS—Shutdown

BASES

BACKGROUND

A description of the Core Spray (CS) System and the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System is provided in the Bases for LCO 3.5.1, "ECCS—Operating."

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated loss of coolant accident (LOCA). The long term cooling analysis following a design basis LOCA (Ref. 1) demonstrates that only one low pressure ECCS injection/spray subsystem is required, post LOCA, to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. It is reasonable to assume, based on engineering judgement, that while in MODES 4 and 5, one low pressure ECCS injection/spray subsystem can maintain adequate reactor vessel water level. To provide redundancy, a minimum of two low pressure ECCS injection/spray subsystems are required to be OPERABLE in MODES 4 and 5.

The low pressure ECCS subsystems satisfy Criterion 3 of the NRC Policy Statement. (10 CFR 50.36 (c)(2)(ii))

LCO

Two low pressure ECCS injection/spray subsystems are required to be OPERABLE. The low pressure ECCS injection/spray subsystems consist of two CS subsystems and two LPCI subsystems. Each CS subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or condensate storage tank (CST) to the reactor pressure vessel (RPV). Each LPCI subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. Only single LPCI pump is required per subsystem because of the larger injection capacity in relation to a CS subsystem. In MODES 4 and 5, the RHR System cross-tie valve is not required to be closed.

(contaminated) (C) (S) [D] (on the CCSTs) (similar) (In addition,) [D] (LPCI) (S) (are) (open)

(continued)

BASES

ACTIONS

C.1, C.2, D.1, D.2, and D.3 (continued)

necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the Surveillance may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

move to previous page

The 4 hour Completion Time to restore at least one low pressure ECCS injection/spray subsystem to OPERABLE status ensures that prompt action will be taken to provide the required cooling capacity or to initiate actions to place the plant in a condition that minimizes any potential fission product release to the environment.

SURVEILLANCE REQUIREMENTS

~~SR 3.5.2.1 and~~ SR 3.5.2.2

above the bottom of the suppression chamber

The minimum water level of 12 ft 2 inches required for the suppression pool is periodically verified to ensure that the suppression pool will provide adequate net positive suction head (NPSH) for the CS System and LPCI subsystem pumps, recirculation volume, and vortex prevention. With the suppression pool water level less than the required limit, all ECCS injection/spray subsystems are inoperable unless they are aligned to an OPERABLE CST.

and LPCI subsystems are

When suppression pool level is < 12 ft 2 inches, the CS System is considered OPERABLE only if it can take suction from the CST, and the CST water level is sufficient to provide the required NPSH for the CS pump. Therefore, a verification that either the suppression pool water level is ≥ 12 ft 2 inches or that CS is aligned to take suction from the CST and the CST contains ≥ 150,000 gallons of water, equivalent to 12 ft, ensures that the CS System can supply at least 150,000 gallons of makeup water to the RPV. The CS suction is uncovered at the 100,000 gallon level. However, as noted, only one required CS subsystem may take credit for the CST option during OPDRVs. During OPDRVs, the volume in the CST may not provide adequate makeup if the RPV were completely drained. Therefore, only one CS subsystem is allowed to use the CST. This ensures the other required ECCS subsystem has adequate makeup volume.

volume

and LPCI pump

and vortex prevention

are

140,000

available

90,000

required low pressure ECCS injection/spray subsystems

and LPCI

S are

low pressure ECCS injection/spray

(continued)

23 ft in both CCSTs with the CCSTs crosstied

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.1.2	Verify drywell-to-suppression chamber bypass leakage is $\leq$ 2% of the acceptable $A/\sqrt{k}$ design value of 0.18 ft^2 at an initial differential pressure of $\geq$ 1.0 psid.	24 months AND ------NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass ---------- 12 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify each suppression pool spray subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.2.4.2	Verify each suppression pool spray nozzle is unobstructed.	10 years

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Suspend CORE ALTERATIONS. AND C.3 Initiate action to suspend OPDRVs.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is $\geq$ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.3	Verify the secondary containment can be maintained $\geq$ 0.25 inch of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate $\leq$ 4000 cfm.	24 months on a STAGGERED TEST BASIS for each SGT subsystem
SR 3.6.4.1.4	Verify all secondary containment equipment hatches are closed and sealed.	24 months

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. Failure to meet air lock leakage limit (SR 3.6.1.2.1) or main steam isolation valve leakage limit (SR 3.6.1.3.10) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program.

As left leakage prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.1.2

Maintaining the pressure suppression function of the primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR measures drywell-to-suppression chamber differential pressure during a 7.5 minute period to ensure that the leakage paths that would bypass the suppression pool are within allowable limits.

Satisfactory performance of this SR can be achieved by establishing a known differential pressure ($\geq$ 1.0 psid) between the drywell and the suppression chamber and verifying that the measured bypass leakage is $\leq$ 2% of the acceptable $A/\sqrt{K}$ design value of 0.18 ft^2 (Ref. 4). The leakage test is performed every 24 months. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed during a unit outage and also in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation, in this event, the Note indicates, increasing the Frequency to once every 12 months is required until the situation is remedied as evidenced by passing two consecutive tests.

REFERENCES

1. UFSAR, Section 6.2.1.

2. UFSAR, Section 15.6.5.

3. 10 CFR 50, Appendix J, Option B.

4. Dresden Station Special Report No. 23, "Information Concerning Dresden Units 2 and 3 Drywell to Torus Vacuum Breakers," April 1973.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.1 (continued)

accident analysis. This is acceptable since the suppression pool spray mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.4.2

This Surveillance is performed every 10 years to verify that the spray nozzles are not obstructed and that spray flow will be provided when required. The 10 year Frequency is adequate to detect degradation in performance due to the passive nozzle design and has been shown to be acceptable through operating experience.

REFERENCES

1. UFSAR, Section 6.2.

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown. C

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

This SR ensures that the secondary containment boundary is sufficiently leak tight to preclude exfiltration under expected wind conditions. The 24 hour Frequency of this SR was developed based on operating experience related to secondary containment vacuum variations during the applicable MODES and the low probability of a DBA occurring.

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal secondary containment vacuum condition.

SR 3.6.4.1.2 and SR 3.6.4.1.4 D

Verifying that one secondary containment access door in each access opening is closed and each equipment hatch is closed and sealed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. In addition, for equipment hatches that are floor plugs, the "sealed" requirement is effectively met by gravity. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. An access opening contains one inner and one outer door. In some cases a secondary containment barrier contains multiple inner or multiple outer doors. For these cases, the access openings share the inner door or the outer door, i.e., the access openings have a common inner door or outer door. The intent is to not breach the secondary containment at any D D D

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.2 and SR 3.6.4.1.4 (continued)

time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times, i.e., all inner doors closed or all outer doors closed. Thus each access opening has one door closed. However, all secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency for SR 3.6.4.1.2 has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door status that are available to the operator. The 24 month Frequency for SR 3.6.4.1.4 is considered adequate in view of the existing administrative controls on equipment hatches.

SR 3.6.4.1.3

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. Each SGT subsystem is designed to maintain the secondary containment at $\geq$ 0.25 inches of vacuum water guage for 1 hour at a flow rate of $\leq$ 4000 cfm. To ensure that all fission products released to the secondary containment are treated, SR 3.6.4.1.3 verifies that a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary can be maintained. When the SGT System is operating as designed, the maintenance of secondary containment pressure cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.3 demonstrates that the pressure in the secondary containment can be maintained $\geq$ 0.25 inches of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate $\leq$ 4000 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. The primary purpose of the SR is to ensure secondary containment boundary integrity. The secondary purpose of the SR is to ensure that the SGT subsystem being tested functions as designed. There is a separate LCO with Surveillance Requirements that serves the primary purpose of ensuring OPERABILITY of the SGT System. This SR need not be performed with each SGT subsystem. The SGT subsystem used for this Surveillance is staggered to ensure that in

(continued)

A.1

3.7 - LIMITING CONDITIONS FOR OPERATION

K. Suppression Chamber

The suppression chamber shall be OPERABLE with:

1. The suppression pool water level between 14' 6.5" and 14' 10.5",

2. A suppression pool maximum average water temperature of ≤95°F during OPERATIONAL MODE(s) 1 or 2, except that the maximum average temperature may be permitted to increase to:

   a. ≤105°F during testing which adds heat to the suppression pool.

   b. ≤110°F with THERMAL POWER ≤1% of RATED THERMAL POWER.

   c. ≤120°F with the main steam line isolation valves closed following a scram.

SR 3.6.1.1.2 3. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1 inch diameter orifice at a differential pressure of 1.0 psid.

A.6

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the suppression pool water level outside the above limits, restore the water level to within the limits

L.1 add proposed ACTION A

4.7 - SURVEILLANCE REQUIREMENTS

K. Suppression Chamber

The suppression chamber shall be demonstrated OPERABLE:

1. By verifying the suppression pool water level to be within the limits at least once per 24 hours.

2. At least once per 24 hours by verifying the suppression pool average water temperature to be ≤95°F, except:

   a. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature to be ≤105°F.

   b. At least once per hour when suppression pool average water temperature is ≥ 95°F, by verifying:

      1) Suppression pool average water temperature to be ≤ 110°F, and

      2) THERMAL POWER to be ≤ 1% of RATED THERMAL POWER after suppression pool average water temperature has exceeded 95°F for more than 24 hours.

   c. At least once per 30 minutes with the main steam isolation valves closed following a scram and suppression pool average water temperature >95°F, by verifying suppression pool average water temperature to be ≤120°F.

D

See ITS 3.6.2.1 and ITS 3.6.2.2

3.7 - LIMITING CONDITIONS FOR OPERATION

~~within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

2. ~~In OPERATIONAL MODE(s) 1 or 2 with the suppression pool average water temperature >95°F, except as permitted above, restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.~~

3. ~~With the suppression pool average water temperature >105°F during testing which adds heat to the suppression pool, except as permitted above, stop all testing which adds heat to the suppression pool and restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.~~

4. ~~With the suppression pool average water temperature >110°F, immediately place the reactor mode switch in the Shutdown position and operate at least one low pressure coolant injection loop in the suppression pool cooling mode.~~

5. ~~With the suppression pool average water temperature >120°F, depressurize the reactor pressure vessel to <150 psig (reactor steam dome pressure) within 12 hours.~~

see ITS 3.6.2.1 and ITS 3.6.2.2

4.7 - SURVEILLANCE REQUIREMENTS

~~3. Deleted.~~

~~4. Deleted.~~

SR 3.6.1.1.2

5. At least once per ~~18~~ 24 months by conducting a drywell to suppression chamber bypass leak test at an initial differential pressure of 1.0 psid and verifying that the measured leakage is within the specified limit. ~~If any drywell to suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission.~~ If two consecutive tests fail to meet the specified limit, a test shall be performed at least every ~~8~~ 12 months until two consecutive tests meet the specified limit, at which time the ~~18~~ 24 month test schedule may be resumed.

LD.1 D L.2 LD.1 C

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.2 The requirement in CTS 4.7.K.5 for the NRC to review the test schedule for subsequent tests if any leak rate test result is not within the required limits has been deleted since the NRC has already approved the test schedule. If one test fails, the current Technical Specifications do not require the test frequency to be changed. The test frequency is only required to be changed if two consecutive tests have failed, as stated in CTS 4.7.K.5. Since the test schedule is already covered by the Technical Specifications, which has been approved by the NRC, there is no reason to have a requirement that the NRC review the test schedule (which will not change from the current test schedule) when one test fails. In addition, a historical review has shown this Surveillance has never failed. Therefore, this change is considered to be acceptable.

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS: 3.6.2.4 - SUPPRESSION POOL SPRAY

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 4.7.L.1 requires verification that each suppression pool spray valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position. The suppression pool spray function is manually actuated (requiring reposition of valves and starting of the LPCI pump by the operator). In the CTS, this is recognized and interpreted that "in the correct position" allows the valves to be in a non-accident position provided they can be realigned to the correct position. In the ITS, the words "in the correct position" mean that the valves must be in the accident position, unless they can be automatically aligned on an accident signal. If so, then they can be in the non-accident position. Thus, for suppression pool spray the additional words "or can be aligned to the correct position" have been added in proposed SR 3.6.2.4.1 to clarify that it is permissible for this systems' valves to be in the non-accident position and still be considered OPERABLE. In addition, since there are no automatic valves for the suppression pool spray mode, the reference to check automatic valves has been deleted. Since these are the current requirements, these changes are considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 A new Surveillance Requirement has been added. This Surveillance Requirement (SR 3.6.2.4.2) verifies each suppression pool spray nozzle is unobstructed every 10 years. This SR is required to ensure that when a suppression pool spray subsystem is required per its design function that it will perform as designed. This SR is an additional restriction on plant operation. D

A.1

ITS 3.6.4.1

3.7 - LIMITING CONDITIONS FOR OPERATION

N. SECONDARY CONTAINMENT INTEGRITY

LCO 3.6.4.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

OPERABLE

A.2

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

ACTION A 1. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODES(s) 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

to OPERABLE status

A.2

ACTION B

2. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

OPERABLE

ACTION C

4.7 - SURVEILLANCE REQUIREMENTS

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

SR 3.6.4.1.1 1. Verifying at least once per 24 hours that the pressure within the secondary containment is ≥0.25 inches of vacuum water gauge.

2. Verifying at least once per 31 days that:

A.3

SR 3.6.4.1.2 a. At least one door in each secondary containment air lock is closed.

b. All secondary containment penetrations[a] not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed.

moved to ITS 3.6.4.2

A.4

LD.1

SR 3.6.4.1.3 3. At least once per 18 months by operating one standby gas treatment subsystem at a flow rate ≤4000 cfm for one hour and maintaining ≥0.25 inches of vacuum water gauge in the secondary containment.

24

on a STAGGERED TEST BASES

M.1

add Proposed SR 3.6.4.1.4

M.2

D

Applicability * When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

a Valves and blind flanges in high-radiation areas may be verified by use of administrative controls. Normally locked or sealed-closed penetrations may be opened intermittently under administrative controls.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 4.7.N.3 requires that one subsystem be tested every 18 months. However, the same SGT subsystem could be tested at each testing occurrence. Proposed SR 3.6.4.1.3 will now require both subsystems be tested in the course of 48 months, as represented by the Staggered Test Basis requirement of the 24 month Frequency. This will ensure each SGT subsystem can maintain the proper vacuum. This is an additional restriction on plant operation.

M.2 A new Surveillance is being added, ITS SR 3.6.4.1.4, which requires all secondary containment equipment hatches to be verified closed and sealed every 24 months. This SR provides adequate assurance that exfiltration from the secondary containment through these hatches will not occur. D

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LD.1 The Frequency for performing CTS 4.7.N.3 has been extended from 18 months to 24 months in proposed SR 3.6.4.1.3 to facilitate a change to the Dresden 2 and 3 refuel cycle from 18 months to 24 months. This surveillance ensures that the Secondary Containment is OPERABLE. The proposed change will allow this Surveillance to extend the Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.B and proposed Specification 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

CTS 4.7.N.3 (ITS SR 3.6.4.1.3) verifies the secondary containment can be maintained at the required vacuum. The purpose of this test is to ensure secondary containment boundary integrity by demonstrating that secondary containment vacuum assumed in the safety analysis can be maintained under design basis conditions. Extending the surveillance interval for this verification of secondary containment integrity is acceptable because secondary containment is maintained at a negative pressure during normal operation, and secondary containment structural integrity is maintained through administrative controls which ensure that no significant changes will be made to the secondary containment structure without proper evaluation. Furthermore, based on engineering judgement, any structural degradation which would result in

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) impacting secondary containment OPERABILITY is not likely to occur during normal plant operation. Any event which would cause significant structural degradation, such as a seismic event would require a plant evaluation.

Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. In addition, the proposed 24 month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

"Specific"

None

RELOCATED SPECIFICATIONS

None

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.7.A>	SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program ~~10 CFR 50, Appendix J, as modified by approved exemptions~~. [1] ~~The leakage rate acceptance criterion is ≤ 1.0 L_a. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are < 0.6 L_a for the Type B and Type C tests, and < 0.75 L_a for the Type A test.~~	~~-----NOTE----- SR 3.0.2 is not applicable ----------~~ In accordance with the Primary Containment Leakage Rate Testing Program ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ [1]
<4.7.K.5>	SR 3.6.1.1.2	Verify drywell to suppression chamber differential pressure does not decrease at a rate > [0.25] inch water gauge per minute tested over a [10] minute period at an initial differential pressure of 1.1 psid. [3] ≥ 1.0 bypass leakage is ≤ 2% of the acceptable $A\sqrt{K}$ design value of 0.18 ft²	24 ~~18~~ months [3] AND -----NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass ---------- 12 ~~18~~ months [3] C D

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.1.1 - PRIMARY CONTAINMENT

1. A 10 CFR 50 Appendix J Testing Program Plan has been added to Section 5.5. The program references the requirements of 10 CFR 50 Appendix J and approved exemptions, therefore, the surveillances have been modified to reference the program. This is consistent with Current Licensing Basis and with TSTF-52.

2. Not used. D

3. The brackets have been removed and the proper plant specific values have been included.

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.7.L> SR 3.6.2.4.1	Verify each RHR suppression pool spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
~~SR 3.6.2.4.2~~	~~Verify each RHR pump develops a flow rate ≥ [400] gpm through the heat exchanger while operating in the suppression pool spray mode.~~	~~In accordance with the Inservice Testing Program or 92 days~~
<DOC M.1> SR 3.6.2.4.2	Verify each suppression pool spray nozzle is unobstructed.	10 years

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.2.4 - SUPPRESSION POOL SPRAY

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. The Dresden 2 and 3 design does not include an automatically actuated Suppression Pool Spray System; the system is entirely manually actuated. Therefore, the word "automatic" has been deleted from the valve position check Surveillance (ITS SR 3.6.2.4.1).

3. Editorial change made to be consistent with other similar specifications.

4. The bracketed requirement has been deleted. The current licensing basis for Dresden 2 and 3 does not require a suppression pool spray flow rate verification.

5. A new Surveillance was added which verifies each suppression pool spray nozzle is unobstructed every 10 years. This Surveillance is required to ensure that when a suppression pool spray subsystem is required per its design function that it will perform as designed. If the spray nozzles are obstructed, then their design function may not be met. D D

<CTS>

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.N Act 2>	C. (continued)	C.2 Suspend CORE ALTERATIONS.	Immediately
		AND	
		C.3 Initiate action to suspend OPDRVs.	Immediately

1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.7.N.1>	SR 3.6.4.1.1	Verify [secondary] containment vacuum is ≥ [0.25] inch of vacuum water gauge.	24 hours
<N.2> 2 4	SR 3.6.4.1.2	Verify all [secondary] containment equipment hatches are closed and sealed.	~~31 days~~ 24 months 2
2 <4.7.N.2>	SR 3.6.4.1.3	Verify ~~each~~ one [secondary] containment access door in each access opening is closed, except when the access opening is being used for entry and exit[, then at least one door shall be closed].	1 31 days TSTF-18
	~~SR 3.6.4.1.4~~	~~Verify each standby gas treatment (SGT) subsystem will draw down the [secondary] containment to ≥ [0.25] inch of vacuum water gauge in ≤ [120] seconds.~~	~~[18] months on a STAGGERED TEST BASIS~~ 3

(continued)

move to proper location

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.4.1 - SECONDARY CONTAINMENT

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. ISTS SR 3.6.4.1.2 verifies all secondary containment equipment hatches are closed and sealed every 31 days. The Surveillance Requirement was not added during the Technical Specification Upgrade Program, in accordance with Amendment 171 and 167 respectively, issued by the NRC on November 27, 1995. However, the SR will be added at a 24 month Frequency (ITS SR 3.6.4.1.4). The following requirements have been revised or renumbered, where applicable, to reflect this change. D D

3. The bracketed Surveillance (ISTS SR 3.6.4.1.4), the drawdown test, has been deleted consistent with the current licensing basis. The analysis does not assume an explicit drawdown time. The subsequent SR has been renumbered to reflect the deletion.

4. ISTS SR 3.6.1.4.5 is a test that ensures the Secondary Containment is Operable; the leak tightness of the Secondary Containment boundary is within the assumptions of the accident analyses. However, it is written in such a manner that it implies that if a SGT subsystem is inoperable, the SR is failed ("Verify each standby gas treatment (SGT) subsystem can..."). As stated above, this is not the intent of the SR. Therefore, to ensure this misinterpretation cannot occur, the SR has been rephrased to more clearly convey the original intent of the SR, to verify the Secondary Containment is Operable. With the new wording, if a SGT subsystem is inoperable, ITS SR 3.6.4.1.3 will still be met and only the SGT System Specification, LCO 3.6.4.3, will be required to be entered. The SR will still ensure each SGT subsystem is used (on a STAGGERED TEST BASIS) to perform the SR. This change is also consistent with TSTF-322.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program ~~10 CFR 50, Appendix J (Ref. 3) as modified by approved exemptions~~. Failure to meet air lock leakage limit ~~testing~~ (SR 3.6.1.2.1), ~~[secondary containment bypass leakage (SR 3.6.1.3.12),] [resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.7),]~~ or main steam isolation valve leakage ~~(SR 3.6.1.3.13)~~ does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of ~~10 CFR 50, Appendix J, as modified by approved exemptions (Ref. 3)~~. As ~~left~~ leakage prior to the first startup after performing a required ~~10 CFR 50, Appendix J,~~ leakage test is required to be ≤ 0.6 L_a for combined Type B and C leakage, and ≤ 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the Primary Containment Leakage Rate Testing Program ~~10 CFR 50, Appendix J (Ref. 3) as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~

SR 3.6.1.1.2

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR measures drywell to suppression chamber differential pressure during a 7.5 ~~[10]~~ minute period to ensure that the leakage paths that would bypass the suppression pool are within allowable limits.

Satisfactory performance of this SR can be achieved by establishing a known differential pressure (≥ 1.0 psid) between the drywell and the suppression chamber and verifying that the measured bypass leakage is ≤ 2% of the acceptable $A\sqrt{K}$ design value of 0.18 ft^2 (Ref. 4) ~~pressure in either the suppression chamber or the drywell does not change by more than [0.25] inch of water per minute over a 10 minute period~~. The leakage test is performed every 24 ~~[18]~~ months ~~[18]~~. The 24 ~~[18]~~ month ~~[1]~~ Frequency was developed

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

considering it is prudent that this Surveillance be performed during a unit outage and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, increasing the Frequency to once every 18 months is required until the situation is remedied as evidenced by passing two consecutive tests.

REFERENCES

1. FSAR, Section 6.2.
2. FSAR, Section 15.7.3B.
3. 10 CFR 50, Appendix J.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.1 (continued)

valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.4.2

Verifying each RHR pump develops a flow rate ≥ [400] gpm while operating in the suppression pool spray mode with flow through the heat exchanger ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by Section XI of the ASME Code (Ref. 2). This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is [in accordance with the Inservice Testing Program, but the Frequency must not exceed 92 days].

REFERENCES

1. FSAR, Section 6.2.
2. ASME, Boiler and Pressure Vessel Code, Section XI.

This Surveillance is performed every 10 years to verify that the spray nozzles are not obstructed and that spray flow will be provided when required. The 10 year Frequency is adequate to detect degradation in performance due to the passive nozzle design and has been shown to be acceptable through operating experience.

BASES

ACTIONS C.1, C.2, and C.3 (continued)

movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

This SR ensures that the [secondary] containment boundary is sufficiently leak tight to preclude exfiltration under expected wind conditions. The 24 hour Frequency of this SR was developed based on operating experience related to [secondary] containment vacuum variations during the applicable MODES and the low probability of a DBA occurring ~~between surveillances~~.

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal [secondary] containment vacuum condition.

[1]

[3] one

SR 3.6.4.1.2 and SR 3.6.4.1.4

Verifying that [secondary] containment equipment hatches ~~and~~ access doors ~~are~~ closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the [secondary] containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining [secondary] containment OPERABILITY requires verifying ~~each~~ door in the access opening is closed, ~~except when the access opening is being used for normal transient entry and exit (then at least one door must remain closed)~~. The 31 day Frequency for ~~these~~ SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door ~~and hatch~~ status that are available to the operator.

[3] ~~one~~ (4) is [3] and each is closed and sealed [1] one TSTF-18 [3] 3.6.4.1.2 INSERT B 3.6.4.1.2 [3]

The 24 month Frequency for SR 3.6.4.1.4 is considered adequate in view of the existing administrative controls on equipment hatches.

(continued)

In addition, for equipment hatches that are floor plugs, the "sealed" requirement is effectively met by gravity. [2]

SURVEILLANCE REQUIREMENTS

----------NOTES----------

1. SR 3.8.1.1 through SR 3.8.1.20 are applicable only to the given unit's AC electrical power sources. D

2. SR 3.8.1.21 is applicable to the opposite unit's AC electrical power sources.

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days C
SR 3.8.1.2	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 2. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.8 must be met. 3. A single test of the common DG at the specified Frequency will satisfy the Surveillance for both units. ---------- Verify each DG starts from standby conditions and achieves steady state voltage ≥ 3952 V and ≤ 4368 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	31 days

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources — Operating

LCO 3.8.4 The following DC electrical power subsystems shall be OPERABLE:

a. Two 250 VDC electrical power subsystems;

b. Division 1 and Division 2 125 VDC electrical power subsystems; and

c. The opposite unit's Division 2 125 VDC electrical power subsystem capable of supporting equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.7.4, "Control Room Emergency Ventilation (CREV) System" (Unit 3 only), LCO 3.7.5, "Control Room Emergency Ventilation Air Conditioning (AC) System" (Unit 3 only), and LCO 3.8.1, "AC Sources — Operating."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One 250 VDC battery inoperable as a result of maintenance or testing.	A.1 Restore 250 VDC battery to OPERABLE status.	Prior to exceeding 7 cumulative days per operating cycle of battery inoperability, on a per battery basis, as a result of maintenance or testing

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources — Shutdown

LCO 3.8.5 One 250 VDC and one 125 VDC electrical power subsystem shall be OPERABLE to support the 250 VDC and one 125 VDC Class 1E electrical power distribution subsystems required by LCO 3.8.8, "Distribution Systems — Shutdown." C D

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 DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more DC electrical power distribution subsystems inoperable.	B.1 Restore DC electrical power distribution subsystems to OPERABLE status.	2 hours AND 16 hours from discovery of failure to meet LCO 3.8.7.a
C. One or more required opposite unit Division 2 AC or DC electrical power distribution subsystems inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.1 when Condition C results in the inoperability of a required offsite circuit. ---------- C.1 Restore required opposite unit Division 2 AC and DC electrical power distribution subsystems to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours
E. Two or more electrical power distribution subsystems inoperable that, in combination, result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

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 Design Basis Accidents 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 satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The DC electrical power subsystems — with: a) the required 250 VDC subsystem consisting of one 250 VDC battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus; and b) the required 125 VDC subsystem consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus — are required to be OPERABLE to support some of the required DC distribution subsystems required OPERABLE

(continued)

BASES

LCO (continued)

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 and inadvertent reactor vessel draindown). The associated alternate 125 VDC electrical power subsystem may be used to satisfy the requirements of the 125 VDC subsystems. C

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:

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 are available;

c. Required features 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.

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

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

(continued)

BASES

ACTIONS (continued)

C.1

With one or more required opposite unit Division 2 AC or DC electrical power distribution subsystems inoperable, the redundant required features of the standby gas treatment (SGT) subsystem may not function if a design basis event were to occur. In addition, Unit 2 and Unit 3 share the single train Control Room Emergency Ventilation (CREV) and the associated Air Conditioning (AC) System. Since these systems are powered only from Unit 2, an inoperable Unit 2 Division 2 AC electrical power distribution subsystem could result in a loss of the CREV System and Control Room Emergency Ventilation AC System functions (for both units).

With a standby gas treatment (SGT) subsystem inoperable, LCO 3.6.4.3 requires restoration of the inoperable SGT subsystem to OPERABLE status in 7 days. Similarly, with the CREV System inoperable, LCO 3.7.4 requires restoration of the inoperable CREV System to OPERABLE status within 7 days. With the Control Room Emergency Ventilation AC System inoperable, LCO 3.7.5 requires restoration of the inoperable Control Room Emergency Ventilation AC System to OPERABLE status in 30 days. Therefore, a 7 day Completion Time is provided to restore the required opposite unit Division 2 AC and DC electrical power subsystems to OPERABLE status. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant system(s) and the low probability of a DBA occurring during this time period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.8.1 be entered and Required Actions taken if the inoperable opposite unit AC electrical power distribution subsystem results in an inoperable required offsite circuit. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to

(continued)

A.1 ITS 3.8.5

A.2 <GENERAL DESCRIPTION>

3.9 - LIMITING CONDITIONS FOR OPERATION

D. D.C. Sources - Shutdown

LCO 3.8.5 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

1. One station 250 volt battery with a full capacity charger.
2. One station 125 volt battery with a full capacity charger.

LA.1 one 250 VDC and one 125 VDC

4.9 - SURVEILLANCE REQUIREMENTS

L.1 add proposed Note

D. D.C. Sources - Shutdown

SR 3.8.5.1 The required batteries and chargers shall be demonstrated OPERABLE[a] per the surveillance requirements in Specification 4.9.C.

M.1 to support the 250 VDC and one 125 VDC Class 1E electrical power distribution subsystems required by LCO 3.8.8, "Distribution Systems – Shutdown."

C C D

APPLICABILITY:

OPERATIONAL MODE(s) 4 and 5, and when handling irradiated fuel in the secondary containment.

M.2 add proposed ACTIONS Note

ACTION:

ACTION A With any of the above required station batteries and/or associated charger(s) inoperable, suspend CORE ALTERATIONS, suspend handling of irradiated fuel in the secondary containment, and suspend operations with a potential for draining the reactor vessel.

add proposed Required Action A.1 L.2

add proposed Required Action A.2.4 M.3

C C

LA.2

a An alternate 125 volt battery shall adhere to these same Surveillance Requirements to be considered OPERABLE, except the Unit 2 total battery terminal voltage on float charge shall be verified weekly as ≥130.2 volts.

SR 3.8.5.1

DISCUSSION OF CHANGES
ITS: 3.8.5 - DC SOURCES — SHUTDOWN

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The ITS present the battery hardware components (battery and charger) in the DC Sources LCO (ITS 3.8.5). The battery cell parameters are presented in a separate LCO (ITS 3.8.6).

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The existing requirement of CTS 3.9.D for one 250 VDC and one 125 VDC electrical power sources to be OPERABLE during shutdown conditions is not specific as to what the sources must be powering. The requirement in ITS LCO 3.8.5 specifies that the sources must support an associated division of the onsite Class 1E DC Electrical Power Distribution System required by LCO 3.8.8, "Distribution Systems — Shutdown." This added restriction conservatively assures that at least the 250 VDC and one 125 VDC electrical power distribution subsystems have an OPERABLE DC source (battery and associated charger) supplying it with power, when required. C D

M.2 CTS 3.9.D, "DC Sources — Shutdown" Actions have been modified by a Note stating that LCO 3.0.3 is not applicable (ITS 3.8.5 ACTIONS Note). 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. This clarification is necessary because defaulting to LCO 3.0.3 during irradiated fuel assembly movement in MODE 1, 2, or 3 would require the reactor to be shutdown, but would not require suspension of movement of irradiated fuel assemblies. Therefore, the proposed Note ensures that proper actions are taken when moving irradiated fuel assemblies in MODE 1, 2, or 3 (i.e., LCO 3.0.3 is not applicable and cannot be used in lieu of suspending fuel movement as required by the ACTIONS of the LCO). This change is also consistent with TSTF-36, Rev. 3.

M.3 In the event the necessary DC sources are not OPERABLE, plant conditions are conservatively restricted in CTS 3.9.D Action (ITS 3.8.5 Required Actions A.2.1, A.2.2, and A.2.3) by suspending CORE ALTERATIONS, irradiated fuel handling, and OPDRVs. However, continued operation without the necessary

TECHNICAL CHANGES - MORE RESTRICTIVE

M.3 (cont'd) DC sources should not be considered acceptable. Therefore, ITS 3.8.5 Required Action A.2.4 is added to commence and continue attempts to restore the necessary DC sources. (Note that if actions are taken in accordance with ITS 3.8.5 Required Action A.1, sufficiently conservative measures are assured by the ACTIONS for the individual components declared inoperable without requiring the efforts to restore the inoperable source.) ITS 3.8.5 Required Action A.2.4 results in an action which does not allow continued operation in the existing plant condition. This has the effect of not allowing MODE changes per LCO 3.0.4. Therefore this existing implicit requirement is explicitly addressed in the ITS 3.8.5 ACTIONS.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details relating to system OPERABILITY in CTS 3.9.D (what constitutes a required DC electrical power source) are proposed to be relocated to the Bases. The details for system OPERABILITY are not necessary in the LCO. The definition of OPERABILITY suffices. ITS LCO 3.8.5 will still require one 250 VDC and one 125 VDC electrical power subsystem to be OPERABLE. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS. C

LA.2 The detail of CTS 4.9.D footnote a that an alternate 125 volt battery shall adhere to these same Surveillance Requirements to be considered OPERABLE is proposed to be relocated to the Bases, in the form of a discussion that states the alternate 125 VDC battery can be used to meet the requirements of the LCO. This requirement is not necessary to ensure the OPERABILITY of the alternate batteries. This requirement, the definition of OPERABILITY, and the proposed Surveillances are sufficient to ensure that the requirement will be met. As such, the relocated detail is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 Three of the DC sources Surveillances required to be performed by CTS 4.9.D (CTS 4.9.C.4, 4.9.C.5, and 4.9.C.6) involve tests that would cause the only required OPERABLE 250 VDC battery to be rendered inoperable. This condition presents a significant risk if an event were to occur during the test. The NRC has previously provided Surveillance exceptions in the Dresden 2 and 3 CTS to avoid a similar condition for the AC sources, but the exceptions have not been applied to DC sources. In an effort to consistently address this concern, proposed SR 3.8.5.1 has a Note that excludes performance requirements of Surveillances that would require the required OPERABLE 250 VDC battery to be rendered inoperable. This allowance does not take exception to the requirement for the battery to be capable of performing the particular function - just to the requirement to demonstrate that capability while that source of power is being relied on to support meeting the LCO. 1 C

L.2 An alternative is proposed in the Dresden 2 and 3 ITS to suspending operations if a DC Source is inoperable, and movement of irradiated fuel assemblies, CORE ALTERATIONS, or OPDRVs are being conducted. The alternative, ITS 3.8.5 Required Action A.1, is to declare the affected feature(s) inoperable, and continue to conduct operations (e.g., OPDRVs), if the affected feature(s) ACTIONS allow. Conservative actions can be assured if the affected feature(s) without the necessary DC power is declared inoperable and the associated ACTIONS of the individual feature(s) taken. These conservative actions are currently approved (or will be approved by the ITS amendment) by the NRC. Therefore, this change is considered acceptable. C

RELOCATED SPECIFICATIONS

None

3 Insert SR Notes

----------NOTES----------

1. SR 3.8.1.1 through SR 3.8.1.20 are applicable only to the given unit's AC electrical power sources. D

2. SR 3.8.1.21 is applicable to the opposite unit's AC electrical power sources.

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.7 <(4.9.A.7)> ----------NOTE---------- All DG starts may be preceded by an engine prelube period. ----------	2. A single test of the common DG at the specified Frequency will satisfy the Surveillance for both units.
Verify each DG starts from standby condition and achieves, in ≤ [12] seconds, voltage ≥ [3740] V and ≤ [4580] V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	184 days
SR 3.8.1.8 <(4.9.A.1.b)> ----------NOTE---------- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. ---------- Verify [automatic/[and] manual] transfer of unit power supply from the normal offsite circuit to the alternate offsite circuit.	18 months

(continued)

:

a. in ≤ [12] seconds, voltage ≥ [3740] V and frequency ≥ 58.8 Hz; and

b. steady state

TSTF-163

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.12 <4.9.A.8.e> <DOC M.7>	
----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. ----------	
Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal each DG auto-starts from standby condition and:	[18 months] 24
a. In ≤ [12] seconds after auto-start and during tests, achieves voltage ≥ [3740] V and ≤ [4580] V;	frequency ≥ [58.8] Hz
b. In ≤ [12] seconds after auto-start and during tests, achieves frequency ≥ [58.8] Hz and ≤ [61.2] Hz; and	Steady state voltage ≥ [3952] V and ≤ [4368] V and
c. Operates for ≥ [5] minutes.	
d. Permanently connected loads remain energized from the offsite power system; and	
e. Emergency loads are energized [or auto-connected through the automatic load sequencer] from the offsite power system.	

(continued)

<CTS>

1 Insert LCO 3.8.4

<3.9.C.1> a. Two 250 VDC electrical power subsystems; D

<3.9.C.2> b. Division 1 and Division 2 125 VDC electrical power subsystems; and

<DOC M1.2> c. The opposite unit Division 2 125 VDC electrical power subsystem capable of supporting equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.7.4, "Control Room Emergency Ventilation (CREV) System" (Unit 3 only), LCO 3.7.5, "Control Room Emergency Ventilation Air Conditioning (AC) System" (Unit 3 only), and LCO 3.8.1, "AC Sources—Operating." D

2 Insert ACTIONS

<3.9.C Act 1> <3.9.C Footnote (b)>	A.	One 250 VDC battery inoperable as a result of maintenance or testing.	A.1 Restore 250 VDC battery to OPERABLE status.	Prior to exceeding 7 cumulative days per operating cycle of battery inoperability, on a per battery basis, as a result of maintenance or testing
<3.9.C Act 1> <3.9.C Footnote (b)>	B.	One 250 VDC battery inoperable, due to the need to replace the battery, as determined by maintenance or testing.	B.1 Restore 250 VDC battery to OPERABLE status.	7 days

all changes are [1] unless otherwise identified

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<3.9.C Footnote (a)> <4.9.C.1>	SR 3.8.4.1	Verify battery terminal voltage is ≥ [120] V on float charge.	7 days
<4.9.C.2>	SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify battery connection resistance is ≤ 1.5E-4 ohm for inter-cell connections, ≤ [1.5E-4 ohm] for inter-rack connections, ≤ [1.5E-4 ohm] for inter-tier connections, and ≤ 1.5E-4 ohm for terminal connections.	92 days
			Insert SR 3.8.4.3 [6]
[6] <4.9.C.3.a>	SR 3.8.4.4	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	[12] 24 months TSTF-38
[6] <4.9.C.3.b>	SR 3.8.4.5	Remove visible corrosion and verify battery cell to cell and terminal connections are [clean and tight, and] coated with anti-corrosion material.	[12] 24 months
[6] <4.9.C.3.c>	SR 3.8.4.6	Verify battery connection resistance is ≤ 1.5E-4 ohm for inter-cell connections, ≤ [1.5E-4 ohm] for inter-rack connections, ≤ [1.5E-4 ohm] for inter-tier connections, and ≤ 1.5E-4 ohm for terminal connections.	[12] 24 months

(continued)

a. ≥ 260.4 VDC for each 250 VDC subsystem;
b. ≥ 125.9 VDC for each 125 VDC subsystem; and
— — — NOTE — — — — — — —
c. Only required to be met when the Unit 2 alternate battery is required to be OPERABLE. ≥ 130.2 VDC for Unit 2 alternate battery.

<CTS>

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

<3.9.D> LCO 3.8.5 ~~DC electrical power subsystems shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems—Shutdown."~~ One 250 VDC and one 125 VDC electrical power subsystem shall be OPERABLE. [2] [TSTF-204] [C] [D]

<Appl 3.9.D> APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the [2] secondary containment.

NOTE
LCO 3.0.3 is not applicable [3]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.9.D Act> <Doc M.1> A. One or more required DC electrical power subsystem[s] inoperable. [2]	A.1 Declare affected required feature(s) inoperable. [TSTF-204] [C]	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment. [2]	Immediately
	AND	
		(continued)

to support the 250 VDC and one 125 VDC Class 1E electrical power distribution subsystems required by LCO 3.8.8, "Distribution Systems-Shutdown" [7] [D]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.8.5 - DC SOURCES — SHUTDOWN

1. The proper LCO number has been provided. This change was necessary due to the deletion of ISTS 3.8.7, "Inverters — Operating" and ISTS 3.8.8, "Inverters — Shutdown."

2. The brackets have been removed and the proper plant specific information/value has been provided.

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. This clarification is necessary because defaulting to LCO 3.0.3 during irradiated fuel assembly movement in Mode 1, 2, or 3 would require the reactor to be shutdown, but would not require suspension of movement of irradiated fuel assemblies. Therefore, the proposed Note ensures that proper actions are taken when moving irradiated fuel assemblies in Mode 1, 2, or 3 (i.e., LCO 3.0.3 is not applicable and cannot be used in lieu of suspending fuel movement as required by the ACTIONS of the LCO). This change is also consistent with TSTF-36, Rev. 4.

4. Due to the Dresden 2 and 3 design (spare battery and charger for the 125 VDC Electrical Power System), individual batteries and battery chargers can be tested without compromising compliance with the requirements of the LCO. Therefore, since the test can be performed without compromising the DC loads, the SRs are not excepted from performance for the 125 VDC electrical power subsystem when the unit is shutdown (per the Note to SR 3.8.5.1). C

5. Editorial change made to match the words in the LCO and ACTION requirements.

6. Change made to be consistent with the Writers Guide.

7. The ISTS LCO, as modified by TSTF-204, is not specific as to what the DC sources must be powering. The LCO has been modified to require each DC source to be powering a DC division required OPERABLE by LCO 3.8.8. D

<CTS>

2

Insert LCO 3.8.7

The following electrical power distribution subsystems shall be OPERABLE:

<3.9.E.1> a. Division 1 and Division 2 AC and DC electrical power distribution subsystems; and

<3.9.E.2> b. The portions of the opposite unit's Division 2 AC and DC electrical power distribution subsystem necessary to support equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.7.4, "Control Room Emergency Ventilation (CREV) System" (Unit 3 only), LCO 3.7.5, "Control Room Emergency Ventilation Air Conditioning (AC) System" (Unit 3 only), and LCO 3.8.1, "AC Sources—Operating."

6

Insert 3.8.7 ACTION C

<DOC M.3>

C.	One or more required opposite unit Division 2 AC or DC electrical power distribution subsystems inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.1 when Condition C results in the inoperability of a required offsite circuit. ----------	
		C.1 Restore required opposite unit Division 2 AC and DC electrical power distribution subsystems to OPERABLE status.	7 days

BASES

LCO (continued)

equipment and interconnecting cabling; and each DG/DD subsystem consisting of one battery bank, one battery charger, and the corresponding control equipment and interconnecting cabling are required to be OPERABLE to support required DC distribution subsystems required OPERABLE by LCO 3.8.10, "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 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:

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 are available;

c. Required features 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.

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

ACTIONS

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

If more than one DC distribution subsystem is required according to LCO 3.8.10, the DC subsystems remaining OPERABLE with one or more DC power sources inoperable may be

(continued)

1

Insert B 3.8.7 ACTION C

C.1

With one or more required opposite unit Division 2 AC or DC electrical power distribution subsystems inoperable, the redundant required features of the standby gas treatment (SGT) subsystem may not function if a design basis event were to occur. In addition, Unit 2 and Unit 3 share the single train Control Room Emergency Ventilation (CREV) and the associated Air Conditioning (AC) System. Since these systems are powered only from Unit 2, an inoperable Unit 2 Division 2 AC electrical power distribution subsystem could result in a loss of the CREV System and Control Room Emergency Ventilation AC System functions (for both units).

With a standby gas treatment (SGT) subsystem inoperable, LCO 3.6.4.3 requires restoration of the inoperable SGT subsystem to OPERABLE status in 7 days. Similarly, with the CREV System inoperable, LCO 3.7.4 requires restoration of the inoperable CREV System to OPERABLE status within 7 days. With the Control Room Emergency Ventilation AC System inoperable, LCO 3.7.5 requires restoration of the inoperable Control Room Emergency Ventilation AC System to OPERABLE status in 30 days. Therefore, a 7 day Completion Time is provided to restore the required opposite unit Division 2 AC and DC electrical power subsystems to OPERABLE status. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant system(s) and the low probability of a DBA occurring during this time period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.8.1 be entered and Required Actions taken if the inoperable opposite unit AC electrical power distribution subsystem results in an inoperable required offsite circuit. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.9.1 - REFUELING EQUIPMENT INTERLOCKS

1. The current wording of ISTS 3.9.1 and the associated Applicability could imply that all the refueling equipment interlocks are required at all times during in-vessel fuel movement. The Current Licensing Basis only requires the interlocks associated with the refuel position, not those associated with other positions of the reactor mode switch, and only when the reactor mode switch is in the refuel position, not when it is in the shutdown position. Therefore, to avoid confusion, the LCO and Applicability have been modified to specifically state that the refueling interlocks are those associated with the refuel position, and that it is applicable when the reactor mode switch is in the refuel position. This change is also consistent with TSTF-232.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Changes have been made consistent with proposed TSTF-225, Revision 1. 1C 1D

3.10 SPECIAL OPERATIONS

3.10.3 Single Control Rod Withdrawal – Cold Shutdown

LCO 3.10.3 The reactor mode switch position specified in Table 1.1-1 for MODE 4 may be changed to include the refuel position, and operation considered not to be in MODE 2, to allow withdrawal of a single control rod, and subsequent removal of the associated control rod drive (CRD) if desired, provided the following requirements are met:

a. All other control rods are fully inserted;

b. 1. LCO 3.9.2, "Refuel Position One-Rod-Out Interlock," and

LCO 3.9.4, "Control Rod Position Indication,"

OR

2. A control rod withdrawal block is inserted; and D

c. 1. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," MODE 5 requirements for Functions 1.a, 1.b, 7.a, 7.b, 11, and 12 of Table 3.3.1.1-1,

LCO 3.3.8.2, "Reactor Protection System (RPS) Electric Power Monitoring," MODE 5 requirements, and

LCO 3.9.5, "Control Rod OPERABILITY – Refueling,"

OR

2. All other control rods in a five by five array centered on the control rod being withdrawn are disarmed; at which time LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," MODE 4 requirements, may be changed to allow the single control rod withdrawn to be assumed to be the highest worth control rod.

APPLICABILITY: MODE 4 with the reactor mode switch in the refuel position.

B 3.10 SPECIAL OPERATIONS

B 3.10.1 Reactor Mode Switch Interlock Testing

BASES

BACKGROUND

The purpose of this Special Operations LCO is to permit operation of the reactor mode switch from one position to another to confirm certain aspects of associated interlocks during periodic tests and calibrations in MODES 3, 4, and 5.

The reactor mode switch is a conveniently located, multiposition, keylock switch provided to select the necessary scram functions for various plant conditions (Ref. 1). The reactor mode switch selects the appropriate trip relays for scram functions and provides appropriate bypasses. The mode switch positions and related scram interlock functions are summarized as follows:

a. Shutdown – Initiates a reactor scram; bypasses main steam line isolation and low turbine condenser vacuum scrams; D

b. Refuel – Selects Neutron Monitoring System (NMS) scram function for low neutron flux level operation (but does not disable the average power range monitor scram); bypasses main steam line isolation and low turbine condenser vacuum scrams; D

c. Startup/Hot Standby – Selects NMS scram function for low neutron flux level operation (intermediate range monitors and average power range monitors); bypasses main steam line isolation and low turbine condenser vacuum scrams; and D

d. Run – Selects NMS scram function for power range operation.

The reactor mode switch also provides interlocks for such functions as control rod blocks, scram discharge volume trip bypass, refueling interlocks, and main steam isolation valve isolations.

APPLICABLE SAFETY ANALYSES

The purpose for reactor mode switch interlock testing is to prevent fuel failure by precluding reactivity excursions or core criticality. The interlock functions of the shutdown

(continued)

<CTS>

3.10 SPECIAL OPERATIONS

3 3

3.10.4 Single Control Rod Withdrawal—Cold Shutdown

<3.10.I> LCO 3.10.4

<T1-2 Footnotes (b) and (e)>
<Doc M.2>
<Doc L.2>
<3.10.A>

The reactor mode switch position specified in Table 1.1-1 for MODE 4 may be changed to include the refuel position, and operation considered not to be in MODE 2, to allow withdrawal of a single control rod, and subsequent removal of the associated control rod drive (CRD) if desired, provided the following requirements are met:

a. All other control rods are fully inserted;

b. 1. LCO 3.9.2, "Refuel Position One-Rod-Out Interlock," and

LCO 3.9.4, "Control Rod Position Indication,"

OR

2. A control rod withdrawal block is inserted; and

c. 1. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," MODE 5 requirements for Functions 1.a, 1.b, 7.a, 7.b, ~~10 and 11~~ 11, and 12 of Table 3.3.1.1-1, and 1 2

LCO 3.9.5, "Control Rod OPERABILITY—Refueling,"

OR

2. All other control rods in a five by five array centered on the control rod being withdrawn are disarmed; at which time LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," MODE 4 requirements, may be changed to allow the single control rod withdrawn to be assumed to be the highest worth control rod.

<Appl 3.10.I> APPLICABILITY: MODE 4 with the reactor mode switch in the refuel position.

<T1-2 Footnote (b)>
<Appl 3.10.A>
<3.10.A Footnote (a)>

LCO 3.3.8.2, "Reactor Protection System (RPS) Electric Power Monitoring," MODE 5 requirements, and

B 3.10 SPECIAL OPERATIONS

5 1 B 3.10.2 Reactor Mode Switch Interlock Testing

BASES

BACKGROUND

The purpose of this Special Operations LCO is to permit operation of the reactor mode switch from one position to another to confirm certain aspects of associated interlocks during periodic tests and calibrations in MODES 3, 4, and 5.

The reactor mode switch is a conveniently located, multiposition, keylock switch provided to select the necessary scram functions for various plant conditions (Ref. 1). The reactor mode switch selects the appropriate trip relays for scram functions and provides appropriate bypasses. The mode switch positions and related scram interlock functions are summarized as follows:

a. Shutdown—Initiates a reactor scram; bypasses main steam line isolation and/reactor high water/level scrams; and low turbine condenser vacuum 1 D

b. Refuel—Selects Neutron Monitoring System (NMS) scram function for low neutron flux level operation (but does not disable the average power range monitor scram); bypasses main steam line isolation and/reactor high water/level scrams; D

c. Startup/Hot Standby—Selects NMS scram function for low neutron flux level operation (intermediate range monitors and average power range monitors); bypasses main steam line isolation and/reactor/high water/level scrams; and 1 D

d. Run—Selects NMS scram function for power range operation.

The reactor mode switch also provides interlocks for such functions as control rod blocks, scram discharge volume trip bypass, refueling interlocks, suppression pool makeup and main steam isolation valve isolations.

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The station manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

5.1.2 A unit supervisor shall be responsible for the control room command function (Since the control room is common to both units, the control room command function for both units can be satisfied by a single unit supervisor). During any absence of the unit supervisor from the control room while the unit is in MODE 1, 2, or 3, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the unit supervisor from the control room while the unit is in MODE 4 or 5 or defueled, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

D

5.2 Organization

5.2.2 Unit Staff (continued)

a. A total of three non-licensed operators for the two units is required in all conditions. At least one of the required non-licensed operators shall be assigned to each unit.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

e. The operations manager or shift operations supervisor shall hold an SRO license.

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the shift manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

DISCUSSION OF CHANGES
ITS: 5.1 - RESPONSIBILITY

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG 1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 A new requirement has been added, ITS 5.1.2, which requires a unit supervisor to be responsible for the control room command function (except during his absence, and then a designated licensed individual). This requirement ensures that an individual is designated to be in command of the control room at all times. This change is a more restrictive change on plant operations. D C

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 6.1.A uses the title "Station Manager." In ITS 5.1.1, this specific title is replaced with the generic title "station manager." The specific title is proposed to be relocated to the Quality Assurance (QA) Manual, which is where the description of this specific title is currently located. The allowance to relocate the specific title out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. The various requirements of the station manager are still retained in the ITS. In addition, the ITS also requires the plant specific titles to be in the QA Manual. Therefore, the relocated specific title is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the QA Manual are controlled by the provisions of 10 CFR 50.54.

LA.2 CTS 6.1.B delineates the responsibility of the Shift Manager for directing and commanding the overall operation of the facility on his shift. This requirement is relocated to the UFSAR. ITS 5.1.2 contains the requirement that a unit supervisor shall be responsible for the control room command function (except during his absence, and then a designated licensed individual). Since ITS 5.1.2 C D

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 (cont'd) provides requirements for the control room command function, inclusion of the detailed responsibilities of the Shift Manager in the ITS is not required to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

"Specific"

None

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS: 5.3 - UNIT STAFF QUALIFICATIONS

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG 1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The details in CTS 6.3 for qualification requirements of the Shift Technical Advisor (STA) position are being deleted. These requirements are adequately addressed in CTS 6.2.C (proposed ITS 5.2.2.f) "specified by the Commission Policy Statement on Engineering Expertise on Shift," and therefore, it is unnecessary to restate the qualification requirements. Since the STA position requirements are retained in proposed ITS 5.2.2.f, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 6.3 uses the plant title "Radiation Protection Manager." In ITS 5.3.1, this specific plant title is replaced with the generic title "radiation protection manager." (The title is still used in ITS 5.3.1 when referring to the Regulatory Guide 1.8 title.) The specific title is proposed to be relocated to the Quality Assurance (QA) Manual, which is where the description of this specific title is currently located. The allowance to relocate the specific title out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. In addition, the ITS also requires the plant specific titles to be in the QA Manual. Therefore, the relocated specific title is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the QA Manual are controlled by the provisions of 10 CFR 50.54.

<CTS>

5.0 ADMINISTRATIVE CONTROLS

2 <6.1> 5.1 Responsibility

station

TSTF-65 1

Reviewer's Note not Shown

<6.1.A> 5.1.1 The [Plant Superintendent] shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

manager

TSTF -65

~~The [Plant Superintendent] or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.~~ 3

<1.A.2> 2 5.1.2 (A) The [Shift Supervisor (SS)] shall be responsible for the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 1, 2, or 3, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 4 or 5, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function. or defueled 4

unit supervisor

(D)

(Since the control room is common to both units, the control room command function for both units can be satisfied by a single unit supervisor) 4

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS 5.1 - RESPONSIBILITY

1. This reviewer's note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet the TSTF-65 allowance. This is not meant to be retained in the final version of the plant specific submittal.

2. The brackets have been removed and the proper plant specific information has been provided.

3. The second paragraph of ISTS 5.1.1, regarding review and approval of tests or experiments is deleted. CTS do not delineate this requirement. | D

4. ISTS 5.1.2 is revised to reflect plant practice. | D

5.2 Organization

⟨6.2.B⟩ 5.2.2 Unit Staff (continued)

⟨6.2.B.1⟩ shall be assigned for each control room from which a reactor is operating in MODES 1, 2, or 3. [5]

[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.]

⟨6.2.B.2⟩ TSTF-258 b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, or 3, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room. [c]

⟨6.2.B.3⟩ b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.b for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. [6] [c] [D] TSTF-258

⟨6.2.B.4⟩ c. A radiation protection [1] Health Physics technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position. TSTF-65 [c] TSTF-258

⟨6.2.B.5⟩ ⟨DOC LA.2⟩ d. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators, and key maintenance personnel). [c]

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed: [1]

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

(continued)

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 5.2 - ORGANIZATION

1. The brackets have been removed and the proper plant specific information has been provided.

2. Typographical/grammatical error corrected.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

5. Editorial changes made for enhanced clarity.

6. Changes have been made to ISTS 5.2.2.a to be consistent with current licensing basis.

7. The referenced requirements are Specifications, not CFR requirements. Therefore, the word "Specifications" has been added to clearly state that "5.2.2.a and 5.2.2.f" are Specifications. 1 D

8. The proper plant specific description of the individual to whom the STA provides technical support has been provided.

9. ISTS 5.2 (Organization) is revised by TSTF-258, Rev. 4. In order to maintain consistency, to the maximum extent practicable, between the Administrative Controls Technical Specifications of the ComEd nuclear stations, the following changes of TSTF-258, Rev. 4, are not incorporated in ITS 5.2: 1 C

   a. ISTS 5.2.2.e contains requirements for control of overtime of the plant staff. These requirements were revised by TSTF-258, Rev. 4. 1 C

   b. ISTS 5.2.2.g contains requirements for the Shift Technical Advisor. The title "Shift Technical Advisor (STA)" was deleted by TSTF-258, Rev. 4. 1 C

   Not incorporating these changes to ISTS 5.2 is consistent with the NRC approved ITS for the ComEd Byron and Braidwood Stations.

TSTF-152

INSERT 5.6.1

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrem and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year. [The initial report shall be submitted by April 30 of the year following initial criticality.]

man 2 1 1 2 3

all changes are [1] unless otherwise identified

TSTF-258 changes not adopted

<CTS>

5.0 ADMINISTRATIVE CONTROLS

<6.12> 5.7 High Radiation Area

at 30cm (12 in.)

radiation protection technicians

<6.12.A> 5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but ≤ 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP) or equivalent document. Individuals qualified in radiation protection procedures (e.g., Health Physics Technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

<6.12.A.1> a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

<6.12.A.2> b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

<6.12.A.3> c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified ~~by the Radiation Protection Manager~~ (or equivalent document) in the RWP (or equivalent document) accessible to personnel.

<6.12.B> 5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels > 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work

Insert 5.7.2

(continued)

ATTACHMENT 2

Revision D to LaSalle County Station, Units 1 and 2
Proposed Improved Technical Specifications Submittal
dated March 3, 2000

Revision D to LaSalle County Station
Improved Technical Specifications Summary of Changes

This attachment provides a brief summary of the changes in Revision D of the proposed Improved Technical Specifications (ITS) submittal for LaSalle County Station, Units 1 and 2. The original Technical Specifications amendment request (i.e., Revision 0) was submitted to the NRC by letter dated March 3, 2000, as revised in Revisions A, B, and C, submitted to the NRC by letters dated June 5, 2000, September 1, 2000, and December 18, 2000, respectively.

Changes committed to based on discussions with the NRC reviewers, a recently approved amendment, minor technical changes, and editorial corrections are included in this revision.

Chapter 1.0

1. Typographical errors in the ISTS markup have been corrected (changed the word "page" to "pages" in the Dose Equivalent I-131 definition and added the word "that" to the ECCS Response Time definition). These changes affect the Improved Standard Technical Specifications (ISTS) markup page 1.1-3.

Section 3.1

1. The Bases for ITS SR 3.1.3.1 have been modified to include the information relocated by ITS 3.1.3 DOC LA.2. This change affects ITS 3.1.3, Bases page B 3.1.3-7 and the ISTS markup page B 3.1-18.

2. The scram time to notch position 45 has been modified to be more consistent with the time allowed in NUREG-1433, and the new time is consistent with the safety analysis. This change affects ITS 3.1.4 page 3.1.4-3 and the ISTS markup page 3.1-15.

3. A typographical error has been corrected in ITS 3.1.6 Condition B and SR 3.1.6.1, to make the words in these requirements consistent with the LCO and Condition A (the word "the" was added). This change affects ITS 3.1.6 page 3.1.6-2 and the ISTS markup page 3.1-20.

4. The Standby Liquid Control System figures were modified to provide more detailed information and to be consistent with the statement in ITS 3.1.7 DOC LA.5. This change affects ITS 3.1.7 pages 3.1.7-4 and 3.1.7-5.

Section 3.3

1. Typographical/editorial corrections have been made to ITS 3.3.1.1 (The addition of the letter "s" to the word Function" in ACTIONS Note 2, the identification of the correct Function number in SR 3.3.1.1.13 Note 2, the use of the term "inches" in lieu of "in" for the Allowable Values of Functions 7.a and 7.b, and the correction of an inequality sign in the ISTS markup only). These changes affect ITS 3.3.1.1 pages 3.3.1.1-1, 3.3.1.1-5, 3.3.1.1-8, and 3.3.1.1-9, and the ISTS markup pages 3.3-1, 3.3-5, and 3.3-9.

2. The term "calendar year" has been changed to "12 months" as requested by the NRC. This change affects ITS 3.3.2.1 page 3.3.2.1-2 and Bases page B 3.3.2.1-7, the Discussion of Changes (DOC) for ITS 3.3.2.1, DOC L.2 (page 5), the ISTS

markup insert page 3.3-19b, the Justification for Deviations (JFD) to ITS 3.3.2.1, JFD 9 (page 1), the ISTS Bases markup insert page B 3.3-52i, and the No Significant Hazards Consideration (NSHC) for ITS 3.3.2.1, NSHC L.2 (page 2).

3. The Frequency for SR 3.3.2.1.5 has been changed from 24 months to 92 days, consistent with the actual trip setpoint methodology for the RBM channels. This change affects ITS 3.3.2.1 page 3.3.2.1-4 and Bases page B 3.3.2.1-11 and the ISTS markup insert page 3.3-19d and Bases insert page B 3.3-52l.

4. A typographical error has been corrected for the ITS 3.3.6.1 Functions 2.c and 2.d (the Allowable Values have been changed from "42" to "42.0"). This change affects ITS 3.3.6.1 page 3.3.6.1-6 and ISTS markup insert page 3.3-57.

5. The change discussed in ITS 3.3.7.1 DOC LB.1 has been reclassified as DOC L.2, as requested by the NRC. This change affects the Current Technical Specifications (CTS) markup for ITS 3.3.7.1, pages 3 of 11 and 8 of 11, the Discussion of Changes for ITS 3.3.7.1, DOC LB.1 (deleted from page 3) and DOC L.2 (pages 7 and 8), and the No Significant Hazards Consideration for ITS 3.3.2.1, NSHC L.2 (page 3).

Section 3.4

1. The time allowed to satisfy the requirements of the LCO in Required Action G.1 has been changed from 12 hours to 24 hours, consistent with the NUREG-1434 and as requested by the NRC. This change affects ITS 3.4.1 page 3.4.1-5 and Bases page B 3.4.1-8, the CTS markup for ITS 3.4.1, pages 1 of 10 and 6 of 10, the Discussion of Changes for ITS 3.4.1, DOC L.1 (page 8), the ISTS markup page 3.4-1 and Bases page B 3.4-4, and the No Significant Hazards Consideration for ITS 3.4.1, NSHC L.1 (page 1).

2. The word "Identify" has been changed to "Verify," as requested by the NRC. This change affects ITS 3.4.5 page 3.4.5-2 and Bases page B 3.4.5-4 and the ISTS markup page 3.4-10 and Bases page B 3.4-25.

3. Typographical errors in SR 3.4.11.1 and SR 3.4.11.2 have been corrected. These changes affect ITS 3.4.11 page 3.4.11-3 and the ISTS markup page 3.4-25.

Section 3.5

1. The words in the LCO Section of the Bases, describing the Note to the LCO, have been modified as requested by the NRC. This change affects ITS 3.5.1 Bases page B 3.5.1-5, ITS 3.5.2 Bases page B 3.5.2-2, and the ISTS Bases markup pages B 3.5-5, insert page B 3.5-5, and insert page B 3.5-14.

Section 3.6

1. The change committed to during discussions with the NRC to resolve a beyond scope issue related to the drywell-to-suppression chamber bypass leakage Surveillance has been made. This change affects ITS 3.6.1.1 page 3.6.1.1-2 and Bases page B 3.6.1.1-5, the CTS markup for ITS 3.6.1.1, pages 3 of 10, 5 of 10,

8 of 10, and 10 of 10, the Discussion of Changes for ITS 3.6.1.1, DOC LA.1 (page 2), and DOC L.3 (page 4), the ISTS markup page 3.6-2 and Bases insert markup page B 3.6-4, and the No Significant Hazards Consideration for ITS 3.6.1.1, NSHC L.3 (page 3).

2. The change committed to during discussions with the NRC to resolve a beyond scope issue related to ITS 3.6.1.1 DOC L.2 has been made. This change affects the Discussion of Changes for ITS 3.6.1.1, DOC L.2 (page 4).

3. The change committed to during discussions with the NRC to resolve Dresden/Quad Cities RAI 3.6.4.1-1 has been made. This change affects ITS 3.6.4.1 page 3.6.4.1-3 and Bases pages B 3.6.4.1-4 and B 3.6.4.1-5, the CTS markup for ITS 3.6.4.1, pages 1 of 2 and 2 of 2, the Discussion of Changes for ITS 3.6.4.1, DOC M.3 (page 2), the ISTS markup page 3.6-45, the Justification for Deviations to ITS 3.6.4.1, JFD 2 (page 1), and the ISTS Bases markup page B 3.6-95.

Section 3.7

1. The term "unit" has been deleted from the LCO and the term "required" has been deleted from SR 3.7.2.1 and the first line of SR 3.7.2.2 for consistency. This change affects ITS 3.7.2 pages 3.7.2-1 and 3.7.2-2 and Bases pages B 3.7.2-1, B 3.7.2-2, and B 3.7.2-4, and the ISTS markup pages, insert page 3.7-7 and page 3.7-8 and Bases markup insert page B 3.7-14 and page B 3.7-16.

2. The Ultimate Heat Sink temperature limit has been modified to be consistent with accident analysis assumptions. This change affects ITS 3.7.3 page 3.7.3-2 and Bases pages B 3.7.3-2 and B 3.7.3-3, the Discussion of Changes for ITS 3.7.3, DOC M.1 (page 1), and the ISTS markup page 3.7-3 and Bases markup pages B 3.7-3, B 3.7-5, and insert page B 3.7-5.

Section 3.8

1. The second part of Condition G has been modified for consistency. This change affects ITS 3.8.1 page 3.8.1-6 and the ISTS markup page 3.8-5.

2. The diesel generator (DG) voltage limits for SR 3.8.1.19 and SR 3.8.1.20 have been changed to be consistent with DG voltage limits for all other SRs. This change affects ITS 3.8.1 pages 3.8.1-18 and 3.8.1-19, the CTS markup for ITS 3.8.1, pages 9 of 28 and 23 of 28, the Discussion of Changes for ITS 3.8.1, DOC M.11 (page 8), and the ISTS markup pages 3.8-17 and 3.8-18.

3. The Bases discussion for the first Applicability Note has been modified as requested by the NRC. This change affects ITS 3.8.1 Bases pages B 3.8.1-6 and B 3.8.1-7 and the ISTS Bases markup page B 3.8-5 and insert page B 3.8-5.

4. A markup error has been corrected in SR 3.8.1.17 (the word "Divisions" has been changed to "Division"). This change affects the ISTS markup page 3.8-16.

5. A change has been made to ITS 3.8.2 Condition D for consistency. This change affects the ISTS 3.8.2 markup insert page 3.8-22.

6. A editorial correction has been made for consistency (addition of the word "oil" into Condition A.2). This change affects ITS 3.8.3 page 3.8.3-1 and the ISTS markup page 3.8-24.

7. An incorrect ISTS 3.8.3 markup page was submitted in Revision C. This page is being resubmitted. This change affects the ISTS 3.8.3 markup page 3.8-26.

8. A change to the description of the modified performance discharge test has been made as requested by the NRC. This change affects ITS 3.8.4 page B 3.8.4-11 and the ISTS Bases markup page B 3.8-58.

9. The IEEE Standard 450 reference date has been change from 1987 to 1995. This change affects ITS 3.8.4 Bases page B 3.8.4-13 and the ISTS Bases markup page B 3.8-60.

10. ITS LCO 3.8.5.a has been modified as requested by the NRC. This change affects ITS 3.8.5 page 3.8.5-1, the CTS markup for ITS 3.8.5, pages 1 of 4 and 3 of 4, the Discussion of Changes for ITS 3.8.5, DOC M.1 (page 2), the ISTS markup insert page 3.8-31, and the Justification for Deviations to ITS 3.8.5, JFD 7 (page 1).

11. ITS LCO 3.8.6 has been modified to be consistent with the requirements of ITS 3.8.4 and ITS 3.8.5 (with respect to the opposite unit requirements). This change affects ITS 3.8.6 page 3.8.6-1 and the ISTS markup page 3.8-33.

12. Editorial corrections have been made to two DOCs in ITS 3.8.6 (deleted the 125 V change description in DOC M.4 and changed the word "increased" to "decreased" in DOC L.7. These changes affect the Discussion of Changes for ITS 3.8.6, DOC M.4 (page 3) and DOC L.7 (page 7).

13. The Bases for the LCO Section of ITS 3.8.7 have been modified as requested by the NRC. This change affects ITS 3.8.7 Bases page B 3.8.7-4, the ISTS Bases markup page B 3.8-83, and the Justification for Deviations to ITS Bases 3.8.7, JFD 10 (deleted from page 1).

Section 3.9

1. A change was made to the JFD as requested by the NRC. This change affects the Justification for Deviations to ITS 3.9.1, JFD 3 (page 1).

Section 3.10

1. Changes have been made based on Amendments 145 and 131 (LaSalle Units 1 and 2, respectively), which modified the License Condition concerning fuel movement with multiple control rods withdrawn. These changes affect ITS 3.10.5 pages 3.10.5-1 and 3.10.5-2 and Bases page B 3.10.5-2, the Discussion of Changes for ITS 3.10.5, DOC M.1 (page 2), the ISTS markup pages 3.10-16 and 3.10-17, the Justification for Deviations to ITS 3.10.5, JFD 2 (page 1), and the ISTS Bases markup page B 3.10-27.

Revision D to LaSalle County Station
Improved Technical Specifications Summary of Changes

Chapter 5.0

1. The change committed to during discussions with the NRC to resolve RAI 5.0-1 has been made. The change affects ITS 5.1 page 5.1-1, the CTS markup for ITS 5.1, pages 1 of 6 through 6 of 6, the Discussion of Changes for ITS 5.1, DOC A.3 (page 1), DOC LA.1 (pages 1 and 2), and DOC LA.2 (page 2), the ISTS markup page 5.0-1, and the Justification for Deviations to ITS 5.1, JFD 3 (page1) and JFD 4 (page 1).

2. A typographical error has been corrected in ITS 5.2.2.b and in the Justification for Deviations to ITS 5.2 (the reference to Specification 5.2.2.g has been changed to 5.2.2.f). In addition, the reference to 5.5.2.a in JFD 6 has been changed to 5.2.2.a. These changes affect ITS 5.2 page 5.2-2, the ISTS markup page 5.0-3, and the Justification for Deviations to ITS 5.2, JFD 6 (page 1).

LaSalle ITS Rev. D Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 1	
CHAPTERS 1.0 AND 2.0 AND SECTION 3.0	
DISCARD	**INSERT**
ISTS markup page 1.1-3	ISTS markup page 1.1-3

VOLUME 2	
SECTIONS 3.1 AND 3.2	
DISCARD	**INSERT**
ITS page 3.1.4-3	ITS page 3.1.4-3
ITS page 3.1.6-2	ITS page 3.1.6-2
ITS pages 3.1.7-4 and 3.1.7-5	ITS pages 3.1.7-4 and 3.1.7-5
ITS Bases page B 3.1.3-7	ITS Bases page B 3.1.3-7
ISTS markup page 3.1-15	ISTS markup page 3.1-15
ISTS markup page 3.1-20	ISTS markup page 3.1-20
ISTS Bases markup page B 3.1-18	ISTS Bases markup page B 3.1-18

VOLUME 3	
SECTION 3.3	
DISCARD	**INSERT**
ITS page 3.3.1.1-1	ITS page 3.3.1.1-1
ITS page 3.3.1.1-5	ITS page 3.3.1.1-5
ITS pages 3.3.1.1-8 and 3.3.1.1-9	ITS pages 3.3.1.1-8 and 3.3.1.1-9
ITS page 3.3.2.1-2	ITS page 3.3.2.1-2
ITS page 3.3.2.1-4	ITS page 3.3.2.1-4
ITS page 3.3.6.1-6	ITS page 3.3.6.1-6
ITS Bases page B 3.3.2.1-7	ITS Bases page B 3.3.2.1-7
ITS Bases page B 3.3.2.1-11	ITS Bases page B 3.3.2.1-11
Discussion of Changes for ITS 3.3.2.1 page 5	Discussion of Changes for ITS 3.3.2.1 page 5
CTS markup for ITS 3.3.7.1 page 3 of 11	CTS markup for ITS 3.3.7.1 page 3 of 11
CTS markup for ITS 3.3.7.1 page 8 of 11	CTS markup for ITS 3.3.7.1 page 8 of 11
Discussion of Changes for ITS 3.3.7.1 pages 3 through 8	Discussion of Changes for ITS 3.3.7.1 pages 3 through 8

VOLUME 4	
SECTION 3.3	
DISCARD	**INSERT**
ISTS markup page 3.3-1	ISTS markup page 3.3-1
ISTS markup page 3.3-5	ISTS markup page 3.3-5
ISTS markup page 3.3-9	ISTS markup page 3.3-9
ISTS markup insert page 3.3-19b	ISTS markup insert page 3.3-19b
ISTS markup insert page 3.3-19d	ISTS markup insert page 3.3-19d
Justification for Deviations to ITS 3.3.2.1 page 1	Justification for Deviations to ITS 3.3.2.1 page 1
ISTS markup insert page 3.3-57	ISTS markup insert page 3.3-57
ISTS Bases markup insert page B 3.3-52i	ISTS Bases markup insert page B 3.3-52i
ISTS Bases markup insert page B 3.3-52l	ISTS Bases markup insert page B 3.3-52l
No Significant Hazards Consideration for ITS 3.3.2.1 page 2	No Significant Hazards Consideration for ITS 3.3.2.1 page 2
No Significant Hazards Consideration for ITS 3.3.7.1 page 3	No Significant Hazards Consideration for ITS 3.3.7.1 page 3

VOLUME 5	
SECTIONS 3.4 AND 3.5	
DISCARD	**INSERT**
ITS page 3.4.1-5	ITS page 3.4.1-5
ITS page 3.4.5-2	ITS page 3.4.5-2
ITS page 3.4.11-3	ITS page 3.4.11-3
ITS Bases page B 3.4.1-8	ITS Bases page B 3.4.1-8
ITS Bases page B 3.4.5-4	ITS Bases pages B 3.4.5-4
CTS markup for ITS 3.4.1 page 1 of 10	CTS markup for ITS 3.4.1 page 1 of 10
CTS markup for ITS 3.4.1 page 6 of 10	CTS markup for ITS 3.4.1 page 6 of 10
Discussion of Changes for ITS 3.4.1 page 8	Discussion of Changes for ITS 3.4.1 page 8
ISTS markup page 3.4-1	ISTS markup page 3.4-1
ISTS markup page 3.4-10	ISTS markup page 3.4-10
ISTS markup page 3.4-25	ISTS markup page 3.4-25
ISTS Bases markup page B 3.4-4	ISTS Bases markup page B 3.4-4
ISTS Bases markup page B 3.4-25	ISTS Bases markup page B 3.4-25
No Significant Hazards Consideration for ITS 3.4.1 page 1	No Significant Hazards Consideration for ITS 3.4.1 page 1
ITS Bases pages B 3.5.1-5 and B 3.5.1-6	ITS Bases pages B 3.5.1-5 and B 3.5.1-6
ITS Bases page B 3.5.2-2	ITS Bases page B 3.5.2-2
ISTS Bases markup page B 3.5-5 and insert page B 3.5-5	ISTS Bases markup page B 3.5-5 and insert page B 3.5-5
ISTS Bases markup insert page B 3.5-14	ISTS Bases markup insert page B 3.5-14

VOLUME 6	
SECTION 3.6	
DISCARD	**INSERT**
ITS page 3.6.1.1-2	ITS page 3.6.1.1-2
ITS page 3.6.4.1-3	ITS page 3.6.4.1-3
ITS Bases page B 3.6.1.1-5	ITS Bases page B 3.6.1.1-5
ITS Bases pages B 3.6.4.1-4 and B 3.6.4.1-5	ITS Bases pages B 3.6.4.1-4 and B 3.6.4.1-5
CTS markup for ITS 3.6.1.1 page 3 of 10	CTS markup for ITS 3.6.1.1 page 3 of 10
CTS markup for ITS 3.6.1.1 page 5 of 10	CTS markup for ITS 3.6.1.1 page 5 of 10
CTS markup for ITS 3.6.1.1 page 8 of 10	CTS markup for ITS 3.6.1.1 page 8 of 10
CTS markup for ITS 3.6.1.1 page 10 of 10	CTS markup for ITS 3.6.1.1 page 10 of 10
Discussion of Changes for ITS 3.6.1.1 pages 2 through 5	Discussion of Changes for ITS 3.6.1.1 pages 2 through 5
CTS markup for ITS 3.6.4.1 pages 1 of 2 and 2 of 2	CTS markup for ITS 3.6.4.1 pages 1 of 2 and 2 of 2
Discussion of Changes for ITS 3.6.4.1 pages 2 and 3	Discussion of Changes for ITS 3.6.4.1 pages 2 and 3

VOLUME 7	
SECTION 3.6	
DISCARD	**INSERT**
ISTS markup page 3.6-2	ISTS markup page 3.6-2
ISTS markup page 3.6-45	ISTS markup page 3.6-45
Justification for Deviations to ITS 3.6.4.1 page 1	Justification for Deviations to ITS 3.6.4.1 page 1
ISTS Bases markup insert page B 3.6-4	ISTS Bases markup insert page B 3.6-4
ISTS Bases markup page B 3.6-95	ISTS Bases markup page B 3.6-95
No Significant Hazards Consideration for ITS 3.6.1.1 page 3	No Significant Hazards Consideration for ITS 3.6.1.1 page 3

VOLUME 8	
SECTION 3.7	
DISCARD	**INSERT**
ITS pages 3.7.2-1 and 3.7.2-2	ITS pages 3.7.2-1 and 3.7.2-2
ITS page 3.7.3-2	ITS page 3.7.3-2
ITS Bases pages B 3.7.2-1 through B 3.7.2-4	ITS Bases pages B 3.7.2-1 through B 3.7.2-4
ITS Bases pages B 3.7.3-2 and B 3.7.3-3	ITS Bases pages B 3.7.3-2 through B 3.7.3-4
Discussion of Changes for ITS 3.7.3 page 1	Discussion of Changes for ITS 3.7.3 page 1
ISTS markup insert page 3.7-7	ISTS markup insert page 3.7-7
ISTS markup page 3.7-8	ISTS markup page 3.7-8
ISTS markup page 3.7-3	ISTS markup page 3.7-3
ISTS Bases markup insert page B 3.7-14	ISTS Bases markup insert page B 3.7-14
ISTS Bases markup page B 3.7-16	ISTS Bases markup page B 3.7-16
ISTS Bases markup page B 3.7-3	ISTS Bases markup page B 3.7-3
ISTS Bases markup page B 3.7-5 and insert page B 3.7-5	ISTS Bases markup page B 3.7-5 and insert page B 3.7-5

VOLUME 9	
SECTION 3.8	
DISCARD	**INSERT**
ITS page 3.8.1-6	ITS page 3.8.1-6
ITS pages 3.8.1-18 and 3.8.1-19	ITS pages 3.8.1-18 and 3.8.1-19
ITS page 3.8.3-1	ITS page 3.8.3-1
ITS pages 3.8.5-1 through 3.8.5-3	ITS pages 3.8.5-1 through 3.8.5-4
ITS page 3.8.6-1	ITS page 3.8.6-1
ITS Bases pages B 3.8.1-6 through B 3.8.1-15	ITS Bases pages B 3.8.1-6 through B 3.8.1-15
ITS Bases pages B 3.8.4-11 through B 3.8.4-13	ITS Bases pages B 3.8.4-11 through B 3.8.4-13
ITS Bases pages B 3.8.7-4 through B 3.8.7-7	ITS Bases pages B 3.8.7-4 through B 3.8.7-7
CTS markup of ITS 3.8.1 page 9 of 28	CTS markup of ITS 3.8.1 page 9 of 28
CTS markup of ITS 3.8.1 page 23 of 28	CTS markup of ITS 3.8.1 page 23 of 28
Discussion of Changes for ITS 3.8.1 page 8	Discussion of Changes for ITS 3.8.1 page 8
CTS markup of ITS 3.8.5 page 1 of 4	CTS markup of ITS 3.8.5 page 1 of 4
CTS markup of ITS 3.8.5 page 3 of 4	CTS markup of ITS 3.8.5 page 3 of 4
Discussion of Changes for ITS 3.8.5 pages 2 through 5	Discussion of Changes for ITS 3.8.5 pages 2 through 5
Discussion of Changes for ITS 3.8.6 page 3	Discussion of Changes for ITS 3.8.6 page 3
Discussion of Changes for ITS 3.8.6 page 7	Discussion of Changes for ITS 3.8.6 page 7
ISTS markup page 3.8-5	ISTS markup page 3.8-5
ISTS markup pages 3.8-16 through 3.8-18	ISTS markup pages 3.8-16 through 3.8-18
ISTS markup insert page 3.8-22	ISTS markup insert page 3.8-22
ISTS markup page 3.8-24	ISTS markup page 3.8-24
ISTS markup page 3.8-26	ISTS markup page 3.8-26
ISTS markup insert page 3.8-31	ISTS markup insert page 3.8-31
Justification for Deviations to ITS 3.8.5 page 1	Justification for Deviations to ITS 3.8.5 page 1
ISTS markup page 3.8-33	ISTS markup page 3.8-33
ISTS Bases markup page B 3.8-5 and insert page B 3.8-5	ISTS Bases markup page B 3.8-5 and insert page B 3.8-5
ISTS Bases markup page B 3.8-58	ISTS Bases markup page B 3.8-58
ISTS Bases markup page B 3.8-60	ISTS Bases markup page B 3.8-60
ISTS Bases markup page B 3.8-83	ISTS Bases markup page B 3.8-83
Justification for Deviations to ITS Bases 3.8.7 page 1	Justification for Deviations to ITS Bases 3.8.7 page 1

VOLUME 10	
SECTIONS 3.9 AND 3.10	
DISCARD	**INSERT**
Justification for Deviations to ITS 3.9.1 page 1	Justification for Deviations to ITS 3.9.1 page 1
ITS pages 3.10.5-1 and 3.10.5-2	ITS pages 3.10.5-1 and 3.10.5-2
ITS Bases page B 3.10.5-2	ITS Bases page B 3.10.5-2
Discussion of Changes for ITS 3.10.5 page 2	Discussion of Changes for ITS 3.10.5 page 2
ISTS markup pages 3.10-16 and 3.10-17	ISTS markup pages 3.10-16 and 3.10-17
Justification for Deviations to ITS 3.10.5 page 1	Justification for Deviations to ITS 3.10.5 page 1
ISTS Bases markup page B 3.10-27	ISTS Bases markup page B 3.10-27

VOLUME 11	
CHAPTERS 4.0 AND 5.0	
DISCARD	**INSERT**
ITS page 5.1-1	ITS page 5.1-1
ITS page 5.2-2	ITS page 5.2-2
CTS markup of ITS 5.1 pages 1 of 4 through 4 of 4	CTS markup of ITS 5.1 pages 1 of 6 through 6 of 6
Discussion of Changes for ITS 5.1 pages 1 and 2	Discussion of Changes for ITS 5.1 pages 1 and 2
ISTS markup page 5.0-1	ISTS markup page 5.0-1
Justification for Deviations to ITS 5.1 page 1	Justification for Deviations to ITS 5.1 page 1
ISTS markup page 5.0-3	ISTS markup page 5.0-3
Justification for Deviations to ITS 5.2 page 1	Justification for Deviations to ITS 5.2 page 1

(CTS)

1.1 Definitions

(1.10)	DOSE EQUIVALENT I-131 (continued)	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, or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity".
(1.12)	EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
(1.13)	END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured ~~[except for the breaker arc suppression time, which is not measured but is validated to conform to the manufacturer's design value]~~.
(1.19)	ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading

(continued)

In lieu of measurement, response time may be verified for selected components provided that the components and method for verification have been previously reviewed and approved by the NRC.

Table 3.1.4-1
Control Rod Scram Times

----------NOTES----------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 05. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[(a)(b)] (seconds) when reactor steam dome pressure ≥ 800 psig
45	0.52
39	0.80
25	1.77
05	3.20

(a) Maximum scram time from fully withdrawn position based on de-energization of scram pilot valve solenoids as time zero.

(b) Scram times as a function of reactor steam dome pressure when < 800 psig are within established limits.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Nine or more OPERABLE control rods not in compliance with the analyzed rod position sequence.	B.1 --------NOTE--------- RWM may be bypassed as allowed by LCO 3.3.2.1. ----------	
	Suspend withdrawal of control rods.	Immediately
	AND	
	B.2 Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify all OPERABLE control rods comply with the analyzed rod position sequence.	24 hours

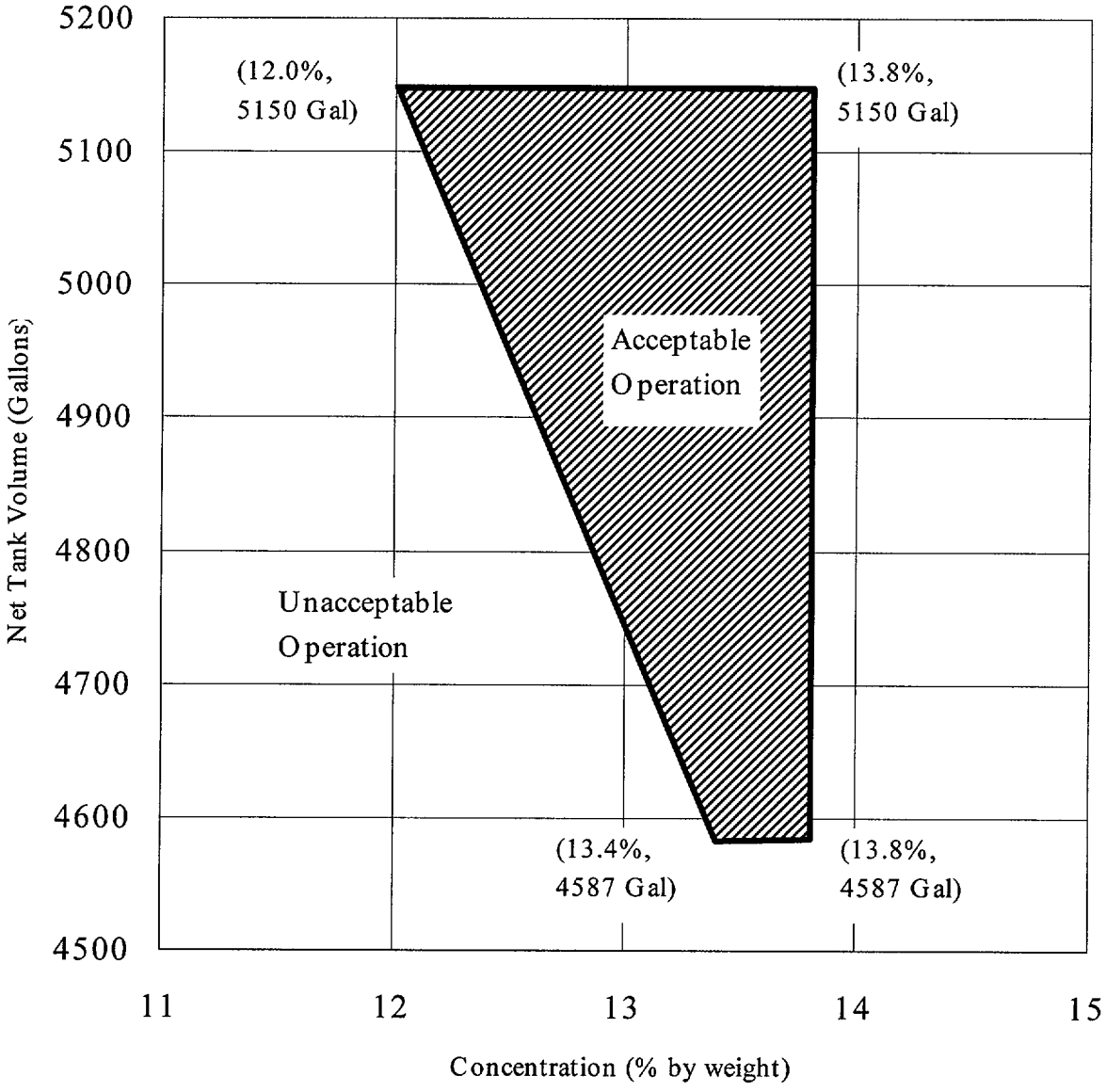

Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume/Concentration Requirements

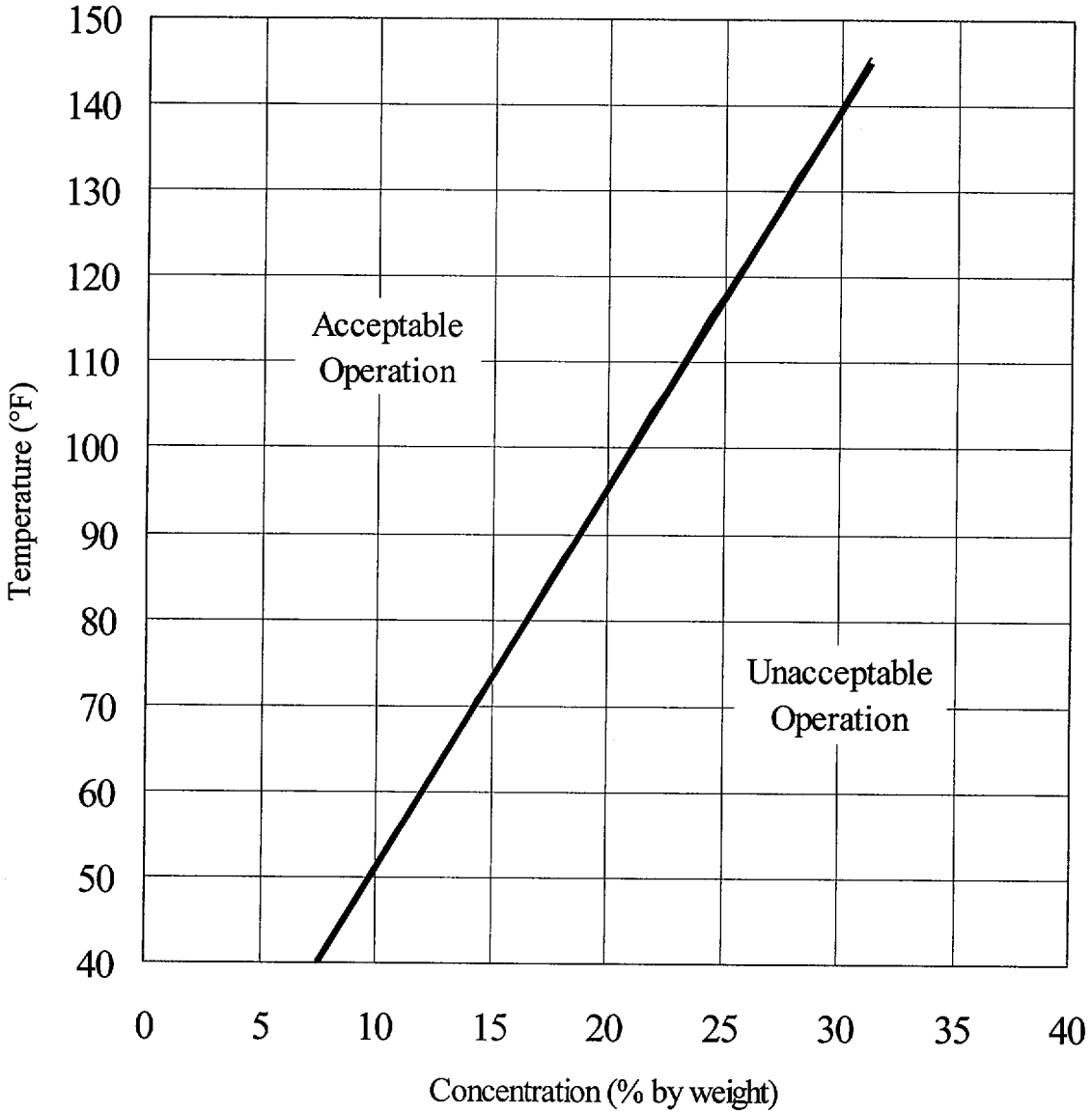

Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Solution Temperature/Concentration Requirements

BASES

ACTIONS (continued)

E.1

If any Required Action and associated Completion Time of Condition A, C, or D are not met or nine or more inoperable control rods exist, 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. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above 10% RTP (i.e., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The position of each control rod must be determined, to ensure adequate information on control rod position is available to the operator for determining control rod OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods by single notch movement to a position with an OPERABLE indicator (full-in, full-out, or numeric indicator) and then returning the control rods by single notch movement to their original position, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room. D D

(continued)

<CTS>

Table 3.1.4-1
Control Rod Scram Times

<DOC M.3>
<LCO 3.1.3.3>
<LCO 3.1.3.4>

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > [ ] seconds to notch position [13]. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES(a)(b) (seconds)	
	REACTOR STEAM DOME PRESSURE ≥ [950] psig	REACTOR STEAM DOME PRESSURE(c) [1050] psig
[13]	[0.30]	[0.31]
[29]	[0.78]	[0.84]
[13]	[1.40]	[1.53]

(a) Maximum scram time from fully withdrawn position based on de-energization of scram pilot valve solenoids as time zero.

(b) Scram times as a function of reactor steam dome pressure when < [950] psig are within established limits.

(c) For intermediate reactor steam dome pressures, the scram time criteria are determined by linear interpolation.

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC M.1> B. Nine or more OPERABLE control rods not in compliance with ~~BPWS~~ the analyzed rod position sequence. [1]	B.1 --------NOTE--------- ~~Affected control rods may be bypassed in RACS in accordance with SR 3.3.2.1.8 for insertion only~~ RWM [2] as allowed by LCO 3.3.2.1 ---------	
	Suspend withdrawal of control rods.	Immediately
	AND	
	B.2 Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<DOC M.1> SR 3.1.6.1	Verify all OPERABLE control rods comply with ~~BPWS~~ the analyzed rod position sequence [1]	24 hours

BASES

ACTIONS (continued)

F.1

If any Required Action and associated Completion Time of Condition A, C, D, or E are not met or nine or more inoperable control rods exist, 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. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above 10% RTP (i.e., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The position of each control rod must be determined, to ensure adequate information on control rod position is available to the operator for determining CRD OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room.

(full-in, full-out, or numeric indicator) and then returning the control rods by single notch movement to their original position

control rod

by single notch movement

SR 3.1.3.2 and SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are

(continued)

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

----------NOTES----------

1. Separate Condition entry is allowed for each channel.
2. When Functions 2.b and 2.c channels are inoperable due to the APRM indication not within limits, entry into associated Conditions and Required Actions may be delayed for up to 2 hours if the APRM is indicating a lower power value than the calculated power, and for up to 12 hours if the APRM is indicating a higher power value than the calculated power. D C

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
	OR		
	A.2	Place associated trip system in trip.	12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	OR		
	B.2	Place one trip system in trip.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 24 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Function 1.a, not required to be performed when entering MODE 2 from MODE 1 until 24 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Verify the APRM Flow Biased Simulated Thermal Power – Upscale time constant is $\leq$ 7 seconds.	24 months
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

(continued)

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	
2. Average Power Range Monitors (continued)						
d. Inop	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.15	NA	
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 1059.0 psig	A
4. Reactor Vessel Water Level – Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 11.0 inches	A
5. Main Steam Isolation Valve – Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 13.7% closed	B
6. Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.93 psig	
7. Scram Discharge Volume Water Level – High						
a. Transmitter/Trip Unit	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 767 ft 8.55 inches elevation	A D
	5[a]	2	H	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 767 ft 8.55 inches elevation	A D

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	
7. Scram Discharge Volume Water Level – High (continued)						
b. Float Switch	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 767 ft 8.55 inches elevation	A D
	5[a]	2	H	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 767 ft 8.55 inches elevation	A D
8. Turbine Stop Valve – Closure	≥ 25% RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 8.9% closed	B
9. Turbine Control Valve Fast Closure, Trip Oil Pressure – Low	≥ 25% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≥ 425.5 psig	A
10. Reactor Mode Switch – Shutdown Position	1,2	2	G	SR 3.3.1.1.12 SR 3.3.1.1.15	NA	
	5[a]	2	H	SR 3.3.1.1.12 SR 3.3.1.1.15	NA	
11. Manual Scram	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.15	NA	
	5[a]	2	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA	

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1.1 Verify ≥ 12 rods withdrawn. OR C.2.1.2 Verify by administrative methods that startup with RWM inoperable has not been performed in the last 12 months. AND C.2.2 Verify movement of control rods is in compliance with analyzed rod position sequence by a second licensed operator or other qualified member of the technical staff.	Immediately Immediately During control rod movement
D. RWM inoperable during reactor shutdown.	D.1 Verify movement of control rods is in compliance with analyzed rod position sequence by a second licensed operator or other qualified member of the technical staff.	During control rod movement

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	----------NOTE---------- Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. ---------- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	----------NOTE---------- Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. ---------- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	----------NOTE---------- Neutron detectors are excluded. ---------- Perform CHANNEL CALIBRATION.	92 days
SR 3.3.2.1.5	----------NOTE---------- Neutron detectors are excluded. ---------- Verify the RBM is not bypassed when THERMAL POWER is ≥ 30% RTP and a peripheral control rod is not selected.	92 days
SR 3.3.2.1.6	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	24 months

(continued)

Table 3.3.6.1-1 (page 1 of 4)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -137.0 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 826.5 psig
c. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 128.0 psid
d. Condenser Vacuum – Low	1,2[(a)], 3[(a)]	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 3.8 inches Hg vacuum
e. Main Steam Line Tunnel Differential Temperature – High	1,2,3	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 66.4°F
f. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.5	NA
2. Primary Containment Isolation					
a. Reactor Vessel Water Level – Low Low, Level 2	1,2,3	2	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ -58.0 inches
b. Drywell Pressure – High	1,2,3	2	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.93 psig
c. Reactor Building Ventilation Exhaust Plenum Radiation-High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 42.0 mR/hr
d. Fuel Pool Ventilation Exhaust Radiation–High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 42.0 mR/hr

(continued)

(a) With any turbine stop valve not closed.

BASES

ACTIONS

B.1 (continued)

The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM during withdrawal of one or more of the first 12 control rods was not performed in the last 12 months. These requirements minimize the number of reactor startups initiated with the RWM inoperable. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2. Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other task qualified member of the technical staff (e.g., shift technical advisor or reactor engineer). D

The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.4 (continued)

The Frequency is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.1.5

The RBM is automatically bypassed when power is below a specified value or if a peripheral control rod is selected. The power level is determined from the APRM signals input to each RBM channel. The automatic bypass setpoint must be verified periodically to be < 30% RTP. In addition, it must also be verified that the RBM is not bypassed when a control rod that is not a peripheral control rod is selected (only one non-peripheral control rod is required to be verified). If any bypass setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the APRM channel can be placed in the conservative condition to enable the RBM. If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8. The 92 day Frequency is based on the actual trip setpoint methodology utilized for these channels.

SR 3.3.2.1.6

The RWM is automatically bypassed when power is above a specified value. The power level is determined from steam flow signal. The automatic bypass setpoint must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

(continued)

TECHNICAL CHANGES - LESS RESTRICTIVE

LF.1 (cont'd) Use of the previously discussed methodologies for determining Allowable Values, instrument setpoints, and analyzing channel/instrument performance ensure that the design basis and associated safety limits will not be exceeded during plant operation. These evaluations, determinations, and analyses now form a portion of the plants design bases.

"Specific"

L.1 CTS Table 4.3.6-1 requires a CHANNEL FUNCTIONAL TEST of the rod block monitor functions to be conducted within 24 hours prior to startup, if not performed within the previous 7 days. This Frequency has not been retained in proposed SR 3.3.2.1.1. The ability of the rod block monitors to perform their function is not impacted by performing a reactor startup. The Frequency defined in proposed SR 3.3.2.1.1 (i.e., 92 days) is sufficient to ensure that the rod block monitors are capable of performing their function. Additionally, the requirements of proposed SR 3.0.4 provide assurance the SR is met within its Frequency prior to entering the MODE or condition requiring OPERABILITY of the equipment.

L.2 With the RWM inoperable prior to a reactor startup, CTS 3.1.4.1 Action c does not allow a startup to commence. Proposed Required Action C.2.1.2 will allow one reactor startup to commence once per 12 months with the RWM inoperable. This change is consistent with the allowance provided by the NRC in their acceptance of NEDE-24011-P-A, Amendment 17. In addition, this change is acceptable since a second licensed operator or other qualified member of the technical staff will verify movement of the control rods is in compliance with the analyzed rod position sequence (Required Action C.2.2). P C

This document provided the requirements for deleting the RSCS System and changing the RWM low power setpoint to 10% RTP. LaSalle 1 and 2 implemented these changes in Amendment 88 (Unit 1) and Amendment 73 (Unit 2); however, the allowance to startup with the RWM inoperable was not provided. In addition, LaSalle 1 and 2 are currently allowed to continue a startup if the RWM becomes inoperable after the first rod is pulled. Proposed Required Action C.2.1.1 will impose additional restrictions in that at least 12 rods must be withdrawn prior to allowing the startup to continue if the RWM becomes inoperable and not crediting this as the one startup with the RWM C

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION

add proposed Required Action A.1

place in trip in 6 hours — L.2

ACTION A

ACTION 70 -

a. With the number of OPERABLE channels per trip system one less than the minimum required, place the inoperable channel in the tripped condition within ~~one~~ 3 6 hour. — L.2 — D

b. With both channels in a trip system inoperable, ~~declare the trip system inoperable, restore the inoperable trip system to OPERABLE status within 7 days,~~ or, within the next ~~8~~ 1 hours, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation. — L.2

ACTION B

c. Otherwise, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation within 1 hour.

add proposed Required Action B.2 — L.2

A.2

Allowable Value

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

SR 3.3.7.1.3

LA.1

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
a. Main Control Room Atmospheric Control System Radiation Monitoring Subsystem	2 per trip system/train (intake)**	1,2,3,5 and *	3.5 mR/hr	0.1 to 10,000 mR/hr	70 A, B

LCO 3.3.7.1

L.1

LF.1

add proposed 3rd and 4th Applicability

A.1

TABLE NOTATIONS

2nd Applicability

*When irradiated fuel is being handled in the secondary containment.

Note to Surveillance Requirements

**A channel may be placed in an inoperable status for up to 6 hours for required surveillance testing without placing the Trip System in the tripped condition, provided at least one other operable channel in the same Trip System is monitoring that Trip Function.

M.2

ACTION STATEMENT

L.2

place in trip in 6 hours

add proposed Required Action A.1

ACTION A

ACTION 70 -

a. With the number of OPERABLE channels per trip system one less than the minimum required, place the inoperable channel in the tripped condition within one hour. 6 5 L.2

b. With both channels in a trip system inoperable, declare the trip system inoperable. Restore the inoperable trip system to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation. 1 L.2

L.2

ACTION B

c. Otherwise, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation within 1 hour.

add proposed Required Action B.2

L.2

D

TECHNICAL CHANGES - MORE RESTRICTIVE

M.2 (cont'd) documented in the NRC Safety Evaluation Report (SER) dated August 2, 1995. The SER concluded that the generic reliability analysis is applicable to LaSalle 1 and 2, and that LaSalle 1 and 2 meets all requirements of the NRC SER accepting the generic reliability analysis.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The measurement range of the Main Control Room Atmospheric Control System Radiation Monitoring System channels in CTS Table 3.3.7.1-1 is proposed to be relocated to the UFSAR. This is a design detail that is not necessary to be included in the Technical Specifications to ensure the OPERABILITY of the CRAF System instrumentation. The OPERABILITY requirements, which include the Allowable Value, are adequately addressed in ITS 3.3.7.1 and the associated Surveillance Requirements. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

LD.1 The Frequency for performing the LOGIC SYSTEM FUNCTIONAL TEST portion of CTS 4.7.2.d.2 (proposed SR 3.3.7.1.4) has been extended from 18 months to 24 months. This SR ensures that CRAF System Instrumentation logic will function as designed to ensure proper response during an analyzed event. The proposed change will allow this Surveillance to extend its Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed SR 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. The CRAF System including the actuating logic is designed to be single failure proof, and therefore, is highly reliable. In addition, major deviations in the instrumentation during the operating cycle will be detected since other surveillances are performed such as

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) the CHANNEL CHECK and CHANNEL FUNCTIONAL TEST (proposed SRs 3.3.7.1.1 and 3.3.7.1.2) at a more frequent basis.

Based on the inherent system and component reliability and the testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical surveillance data also demonstrated that there are no failures that would invalidate this conclusion. In addition, the proposed 24-month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months:

"Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Based on the above discussion, the impact, if any, of this change on system availability is minimal.

LE.1 The Frequency for performing the CHANNEL CALIBRATION of CTS 4.3.7.1 as specified in Table 4.3.7.1-1 (proposed SR 3.3.7.1.3) has been extended from 18 months to 24 months. The subject SR ensures that the CRAF control room air intake radiation monitors will function as designed during an analyzed event. The proposed change will allow these Surveillances to extend their Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed SR 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Extending the SR Frequency is acceptable because the isolation initiation logic is designed to be single failure proof, and therefore, is highly reliable.

TECHNICAL CHANGES - LESS RESTRICTIVE

LE.1 (cont'd) **Instrumentation 1:** Main Control Room Atmospheric Control System Radiation Monitor

This function is performed by General Atomics RP-1A radiation monitoring system and Tracor WESTRONICS M11E recorder. These instruments were evaluated utilizing a qualitative analysis (i.e., engineering judgment). The results of the analysis support a 24 month fuel cycle surveillance interval extension.

A review of the surveillance test history was performed to validate the above conclusion. This review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact, if any, on system availability is minimal from a change to a 24-month surveillance frequency. In addition, the proposed 24-month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

LF.1 This change revises the Current Technical Specifications (CTS) Allowable Values to the Improved Technical Specifications (ITS) Allowable Values. ITS Section 3.3 reflects Allowable Values consistent with the philosophy of BWR ISTS, NUREG-1434, Rev. 1. These Allowable Values have been established consistent with the methods described in ComEd's Instrument Setpoint Methodology (Nuclear Engineering Standard NES-EIC-20.04, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy"). For most cases, the Allowable Value determinations were calculated using plant specific operating and surveillance trend data or an allowance as provided for by the Instrument Setpoint Methodology. For all other cases, vendor documented performance specifications for drift were used. The Allowable Value verification used actual plant operating and surveillance trend information to ensure the validity of the developed Allowable Value. All changes to safety analysis limits applied in the methodologies were evaluated and confirmed as ensuring safety analysis licensing acceptance limits are maintained. All design limits applied in the methodologies were confirmed as ensuring that applicable design requirements of the associated systems and equipment are maintained. The methodologies used have been compared with the guidance of ANSI/ISA S67.04-Part I-1994 and ANSI/ISA RP67.04-Part II-1994. Plant calibration procedures will ensure that the assumptions regarding calibration accuracy, measurement and test equipment accuracy, and setting tolerance are maintained. Setpoints for each design or safety analysis limit have been established by accounting for the applicable instrument accuracy, calibration and drift uncertainties, environmental effects, power supply fluctuations, as well as uncertainties related to process and primary element measurement accuracy using the Instrument Setpoint Methodology. The Allowable Values have been established from each design or safety analysis limit

TECHNICAL CHANGES - LESS RESTRICTIVE

LF.1 (cont'd) by combining the errors associated with channel/instrument calibration (e.g., device accuracy, setting tolerance, and drift) with the calculated Nominal Trip Setpoint also using the Instrument Setpoint Methodology.

Additionally, each applicable channel/instrument has been evaluated and analyzed to support a fuel cycle extension to a 24 month interval. These evaluations and analyses have been performed utilizing the guidance provided in EPRI TR-103335, "Guidelines for Instrument Calibration Extension/Reduction Programs, Revision 1. The EPRI guidance was used to demonstrate that the data collected by the operating plant (from surveillance testing) has remained acceptable and reasonable with regard to the manufacturers design specifications.

Use of the previously discussed methodologies for determining Allowable Values, instrument setpoints, and analyzing channel/instrument performance ensure that the design basis and associated safety limits will not be exceeded during plant operation. These evaluations, determinations, and analyses now form a portion of the plants design bases.

"Specific"

L.1 The Applicability of CTS 3/4.7.2 and CTS 3.3.7.1 (Tables 3.3.7.1-1 and 4.3.7.1-1) for the main control room atmospheric control system radiation monitoring subsystem (changed to Control Room Area Filtration System Instrumentation in Discussion of Change A.2) is revised from Operational Conditions 1, 2, 3, 4, and 5 (the MODE 4 applicability only applies to CTS 3/4.7.2), and when irradiated fuel is being handled in secondary containment to MODES 1, 2, and 3, during movement of irradiated fuel assemblies in secondary containment, during CORE ALTERATIONS, and during operations with the potential for draining the reactor vessel (OPDRVs) in ITS 3.3.7.1, Control Room Area Filtration (CRAF) System Instrumentation. The CRAF System is required to be OPERABLE to control operator radiation exposure during and following a design basis accident, since a design basis accident could lead to a fission product release. When the plant is in MODE 4 or 5, the probability and consequences of a design basis accident are reduced due to the temperature and pressure limitations in these MODES. However, in MODE 4 or 5, activities are conducted for which significant releases of radioactivity are postulated. Therefore, the CRAF System (and the associated supporting initiation instrumentation) is only required to be OPERABLE in MODE 4 or 5, when activities are in progress which could, if an event occurs, result is significant releases of radioactivity (during movement of irradiated fuel assemblies in secondary containment, during CORE ALTERATIONS, or during OPDRVs). This change alters the CTS 3.3.7.1 MODE 4 and 5 Applicability to

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) only include these activities. This is considered acceptable since ITS 3.3.7.1 requires the CRAF System instrumentation to be OPERABLE when it is required to mitigate postulated events in MODE 4 or 5. The ITS 3.3.7.1 Applicability maintains and adds situations for which significant releases of radioactivity are postulated while the plant is in MODE 4 or 5. In addition, the change to the Applicability is consistent with the intent of CTS 3/4.7.2, Control Room Emergency Ventilation System ACTIONS (in MODE 4 and 5 with two subsystems inoperable, the CTS ACTIONS require suspension of those activities for which significant releases of radioactivity are postulated). This change allows operations that do not have a potential for a significant radioactive release to be performed without requiring the CRAF System (and its associated supporting initiation instrumentation) to be OPERABLE and provides additional scheduling flexibility during plant refueling outages.

L.2 With two channels in a trip system inoperable, CTS Table 3.3.7.1-1 Action 70.b requires the trip system to be declared inoperable, and to restore the trip system within 7 days. If not restored, the CRAF System must be placed in the pressurization mode within the next 6 hours. Thus, the CTS ensures the CRAF System is in the pressurization mode within 7 days, 6 hours (total time) after two channels in a trip system are inoperable. ITS 3.3.7.1 ACTION A (6 hours to trip a channel), ITS 3.3.7.1 ACTION B (1 hour to declare the CRAF subsystem inoperable), ITS 3.7.4 ACTION A (7 days to restore an inoperable CRAF subsystem), and ITS 3.7.4 ACTION C (place CRAF subsystem in the pressurization mode immediately), provide a total time of 7 days, 7 hours before the CRAF subsystem must be in the pressurization mode. Thus, the ITS allows an additional hour for this final condition. With one channel in a trip system inoperable, CTS Table 3.3.7.1-1 Action 70.a requires the channel to be placed in trip within 1 hour (changed to 6 hours as described below in the second paragraph). If the channel is not tripped, CTS Table 3.3.7.1-1 Action 70.c would require the CRAF System to be placed in operation within 1 hour. ITS 3.3.7.1 ACTION B provides an additional allowance to declare the CRAF subsystem inoperable within 1 hour (Required Action B.2) when a channel is not tripped (as required by CTS Table 3.3.7.1-1 Action 70.a). Once declared inoperable, ITS 3.7.4 ACTION A will allow a 7 day restoration time for the CRAF subsystem. If not restored, ITS 3.7.4 ACTION C will then require the CRAF subsystem to be placed in the pressurization mode immediately. Thus, the ITS allows an additional 7 days for this final condition. The alternative action to declare the associated equipment inoperable and take the additional 1 hour or 7 days, as applicable, is acceptable since the associated CRAF System Specification (ITS 3.7.4) will provide appropriate actions that are identical to actions taken when a CRAF subsystem is inoperable for reasons other than inoperable instrumentation.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 (cont'd) In addition, CTS Table 3.3.7.1-1 Action 70.a requires an inoperable channel to be placed in trip within one hour, if the number of OPERABLE channels per trip system is one less than the minimum required. When one channel is inoperable, the associated trip system cannot actuate the associated CRAF subsystem, since both channels in a trip system must trip to actuate the trip system (i.e., each trip system is a two-out-of-two logic for the radiation monitors). As described above in the first paragraph, when both channels in a trip system are inoperable, the total time allowed by CTS Table 3.3.7.1-1 Action 70.b to be place the CRAF subsystem in the pressurization mode is 7 days, 6 hours (changed to 7 days, 7 hours as described in the first paragraph of this Discussion of Change). Since each trip system is a two-out-of-two logic, one channel inoperable is functionally equivalent to two channels inoperable. That is, with one channel in a trip system inoperable, the trip system cannot actuate the associated CRAF subsystem, identical to the results when two channels in a trip system are inoperable. Therefore, the total time allowed to place the CRAF subsystem in the pressurization mode when one channel in a trip system is inoperable has been changed to 7 days, 7 hours (ITS 3.3.7.1 Required Actions A.2, B.1 and B.2 and ITS 3.7.4 ACTIONS A and C), consistent with the time provided for when two channels in a trip system are inoperable. However, this out of service time is only acceptable provided the radiation monitors are still maintaining CRAF subsystem initiation capability (ITS 3.3.7.1 Required Action A.1). The CRAF System instrumentation is considered to be maintaining subsystem initiation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate an initiation signal on a valid air intake high radiation signal. D

RELOCATED SPECIFICATIONS

None

<CTS>

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

<LCO 3.3.1> LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

<APPL 3.3.1> APPLICABILITY: According to Table 3.3.1.1-1.

<Table 4.3.1.1-1 footnote (d)>

ACTIONS

<DOC A.2>

----------NOTE----------
1. Separate Condition entry is allowed for each channel.
---------- 7

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.3.1 Act a> <3.3.1 Act b> <3.3.1 Act b.3>	A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 Place associated trip system in trip.	12 hours 12 hours
<3.3.1 Act b> <3.3.1 Act b.2>	B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours
<3.3.1 Act b> <3.3.1 Act b.1>	C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

7

2. When Functions 2.b and 2.c channels are inoperable due to the APRM indication not within limits, entry into associated Conditions and Required Actions may be delayed for up to 2 hours if the APRM is indicating a lower power value than the calculated power, and for up to 12 hours if the APRM is indicating a higher power value than the calculated power. D C

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
<Table 4.3.1.1-1>	SR 3.3.1.1.11	----------NOTES---------- 1. Neutron detectors are excluded. 2. For ~~function~~ [9] 2.a, not required to be performed when entering MODE 2 from MODE 1 until ~~12~~ 24 hours after entering MODE 2. [4] ---------- Perform CHANNEL CALIBRATION.	184 days
<Table 4.3.1.1-1>	SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	~~18~~ 24 months [1]
<Table 4.3.1.1-1> <Table 4.3.1.1-1, footnote (a)> <Table 4.3.1.1-1, footnote *>	SR 3.3.1.1.13	----------NOTES---------- 1. Neutron detectors are excluded. 2. For ~~function~~ [9] ~~1~~ .a, not required to be performed when entering MODE 2 from MODE 1 until ~~12~~ 24 hours after entering MODE 2. [4] ---------- Perform CHANNEL CALIBRATION.	~~18~~ 24 months [1]
<Table 4.3.1.1-1> <Table 4.3.1.1-1, footnote (h)>	SR 3.3.1.1.14	Verify the APRM Flow Biased Simulated Thermal Power ~~High~~ Upscale time constant is ≤ ~~7~~ 1 seconds. [5]	~~18~~ 24 months [1]
<4.3.1.2>	SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	~~18~~ 24 months [1]

(continued)

<ETS>

<Table 3.3.1-1>
<Table 4.3.1.1-1>
<Table 2.2.1-1>

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7-2 8. Scram Discharge Volume Water Level – High					
a. Transmitter/Trip Unit	1,2	~~2~~ 1	G-2 ~~F~~	~~SR 3.3.1.1.1~~ SR 3.3.1.1.9 ~~SR 3.3.1.1.10~~ SR 3.3.1.1.13 SR 3.3.1.1.15	≤ ~~[67]~~ % of full scale
	5(a)	~~2~~ 1	H-2 ~~F~~	~~SR 3.3.1.1.1~~ SR 3.3.1.1.9 ~~SR 3.3.1.1.10~~ SR 3.3.1.1.13 SR 3.3.1.1.15	≤ ~~[67]~~ % of full scale
b. Float Switch	1,2	~~2~~ 1	G-2 ~~F~~	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ ~~[65]~~ inches
	5(a)	~~2~~ 1	H-2 ~~F~~	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ ~~[65]~~ inches
8-2 ~~9.~~ Turbine Stop Valve Closure ~~Trip Oil Pressure – Low~~	≥ ~~[45]~~ 25 % RTP	~~4~~ 1	E	SR 3.3.1.1.9 ~~SR 3.3.1.1.10~~ SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	~~≥ [57] psig~~ ≤ 8.9% closed
9-2 ~~10.~~ Turbine Control Valve Fast Closure, Trip Oil Pressure – Low	≥ ~~[40]~~ 25 % RTP	~~2~~ 1	E	SR 3.3.1.1.9 ~~SR 3.3.1.1.10~~ SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≥ ~~[42]~~ 425.5 psig
10-2 ~~11.~~ Reactor Mode Switch – Shutdown Position	1,2	~~2~~ 1	G-2 ~~F~~	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
	5(a)	~~2~~ 1	H-2 ~~F~~	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
11-2 ~~12.~~ Manual Scram	1,2	~~2~~ 1	G-2 ~~F~~	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	~~2~~ 1	H-2 ~~F~~	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.1.4.1 Acta> C. (continued)	C.2.1.1 Verify ≥ 12 rods withdrawn.	Immediately
	OR	
	C.2.1.2 Verify by administrative methods that startup with RWM inoperable has not been performed in the last ~~calendar year~~.	Immediately 12 months [9] [D]
	AND	
	C.2.2 Verify movement of control rods is in compliance with ~~banked position withdrawal sequence (BPWS)~~ analyzed rod position sequence [6] by a second licensed operator or other qualified member of the technical staff.	During control rod movement
<3.1.4.1 Acta> D. RWM inoperable during reactor shutdown.	D.1 Verify movement of control rods is in ~~accordance~~ compliance [2] with ~~BPWS~~ analyzed rod position sequence [6] by a second licensed operator or other qualified member of the technical staff.	During control rod movement

(continued)

Insert Page 3.3-19b

<CTS> INSERT BWR/4 ISTS 3.3.2.1 (CONTINUED) [1]

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
<Appl 3.1.4.1, footnote *> <4.1.4.1.a> <4.1.4.1b>	SR 3.3.2.1.2	----------NOTE---------- Not required to be performed until 1 hour after any control rod is withdrawn at ≤ ~~{10}~~% RTP in MODE 2. ---------- Perform CHANNEL FUNCTIONAL TEST.	~~{92}~~ days [3]
<4.1.4.1a> <4.1.4.1c>	SR 3.3.2.1.3	----------NOTE---------- Not required to be performed until 1 hour after THERMAL POWER is ≤ ~~{10}~~% RTP in MODE 1. ---------- Perform CHANNEL FUNCTIONAL TEST.	~~{92}~~ days [3]
<DOC M.2>	SR 3.3.2.1.4 [4] (5)	----------NOTE---------- Neutron detectors are excluded. ---------- [3] Verify the RBM ~~is not bypassed when THERMAL POWER is ≥ 30% RTP and a peripheral control rod is not selected.~~ [5] ~~a. Low Power Range—Upscale Function is not bypassed when THERMAL POWER is ≥ 29% and ≤ 64% RTP. b. Intermediate Power Range—Upscale Function is not bypassed when THERMAL POWER is > 64% and ≤ 84% RTP. c. High Power Range—Upscale Function is not bypassed when THERMAL POWER is > 84% RTP.~~	~~{18}~~ months (92 days) [3]
<DOC M.4>	SR 3.3.2.1.5 [4] (6)	Verify the RWM is not bypassed when THERMAL POWER is ≤ ~~{10}~~% RTP.	~~{18}~~ months (24) [3]

(continued)

<Insert SR 3.3.2.1.4 from next page> [4]

Insert Page 3.3-19d

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.3.2.1 - CONTROL ROD BLOCK INSTRUMENTATION

1. The BWR/6 ISTS 3.3.2.1 has been deleted and, in its place, the BWR/4 ISTS 3.3.2.1 (from NUREG-1433, Rev. 1) has been used since the LaSalle 1 and 2 design is similar to the BWR/4 design with regards to the control rod block instrumentation. Any deviations from the BWR/4 ISTS are discussed below.

2. Editorial change made to be consistent with Required Action C.2.2.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. The Surveillances have been placed in the proper order, based on decreasing Frequency, consistent with the Writer's Guide convention. The change from the ISTS order was necessary due to the change in Frequency (to be consistent with current licensing basis) for ISTS SR 3.3.2.1.7 (ITS SR 3.3.2.1.4). The requirements have been renumbered, where applicable, to reflect their new order.

5. ISTS SR 3.3.2.1.4 and ISTS Table 3.3.2.1-1, Note (a) have been modified and ISTS Table 3.3.2.1-1, Functions 1.b, 1.c, and 1.f, including Notes (b), (c), (d), and (e) have been deleted to be consistent with the LaSalle 1 and 2 RBM design. The RBM design in the ISTS is based on a "Post-ARTS" RBM design. LaSalle 1 and 2 has not installed the "ARTS" RBM modification. In addition, the requirements have been renumbered, where applicable, to reflect the deletions.

6. The proper LaSalle 1 and 2 plant specific nomenclature/value/design requirements have been provided.

7. A new Surveillance Requirement has been added, ITS SR 3.3.2.1.9. This Surveillance Requirement is similar to the BWR/6 ISTS SR 3.3.2.1.9, since the Rod Worth Minimizer at LaSalle 1 and 2 includes the capability to bypass individual control rods. Therefore, the BWR/6 ISTS SR 3.3.2.1.9 is used and is modified to reflect the LaSalle 1 and 2 current licensing basis.

8. A new Specification has been added, ITS 3.3.2.2. This Specification is from the BWR/4 ISTS (NUREG-1433 ISTS 3.3.2.2), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to the Feedwater System and Main Turbine High Water Level Trip Instrumentation. Therefore, the BWR/4 Specification is used and any deviations from the BWR/4 ISTS are discussed in the Justification for Deviations for ITS: 3.3.2.2.

9. Required Action C.2.1.2 has been modified to be consistent with the Bases.

Insert 2.d, 2.e, and 2.f

d.	Fuel Pool Ventilation Exhaust Radiation—High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 42.0 mR/hr	A D
e.	Reactor Vessel Water Level—Low Low Low, Level 1	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ -137.0 inches	A
f.	Reactor Vessel Water Level—Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 11.0 inches	A

BASES

ACTIONS

A.1 (continued)

reason, Required Action A.1 requires restoration of the inoperable channel to OPERABLE status. The Completion Time of 24 hours is based on the low probability of an event occurring coincident with a failure in the remaining OPERABLE channel.

B.1

If Required Action A.1 is not met and the associated Completion Time has expired, the inoperable channel must be placed in trip within 1 hour. If both RBM channels are inoperable, the RBM is not capable of performing its intended function; thus, one channel must also be placed in trip. This initiates a control rod withdrawal block, thereby ensuring that the RBM function is met.

The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

5 These requirements minimize the number of reactor startups initiated with the RWM inoperable.

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM was not performed in the last 12 months. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2. Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff.

3 during withdrawal of one or more of the first 12 control rods

D

2 task

2 (e.g., shift technical advisor or reactor engineer)

(continued)

Insert Page B 3.3-52i

1 INSERT BWR/4 ISTS B 3.3.2.1 (continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

the prescribed sequence and verifying a control rod block occurs. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and entry into MODE 1 when THERMAL POWER is ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

2 and by verifying proper annunciation of the selection error of at least one out-of-sequence control rod

3 at ≤ 10% RTP

The Note to SR 3.3.2.1.2

5 and 3

3 on a Startup and entry into MODE 2 concurrent with a power reduction to ≤ 10% RTP during a shutdown

C

6 INSERT SR 3.3.2.1.4 from pages B 3.3-53 and B 3.3-54

Insert SR 3.3.2.1.5 2

SR 3.3.2.1.4 5 6

2 INSERT SR 3.3.2.1.5

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1, each within a specific power range. The power at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8. The 18 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

bypass

APRM 2

2 to enable the RBM

92 day 9

D

SR 3.3.2.1.5 6 6

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass

2

(continued)

The Note to SR 3.3.2.1.3 allows a THERMAL POWER reduction to ≤ 10% RTP in MODE 1 to perform the required Surveillance if the 92 day Frequency is Not met per SR 3.0.2. 3 C

Insert Page B 3.3-52L

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.3.2.1 - CONTROL ROD BLOCK INSTRUMENTATION

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The RWM is not assumed to be an initiator of any analyzed event. This change will allow one reactor startup each 12 months to commence with the RWM inoperable. However, the change will also ensure that the RWM is maintained OPERABLE through at least the withdrawal of the first 12 control rods instead of through just the first control rod. In addition, reactor startup with the RWM inoperable can only commence if the rod pattern is verified by two qualified personnel. Therefore, this change does not significantly increase the probability or consequences of an accident previously analyzed.

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

   The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change does not introduce a new mode of plant operation and does not require physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   Allowing the reactor startup to commence with RWM inoperable one time each 12 months is acceptable based on the compensatory measure provided. Two qualified personnel will be required to verify that movement of each control rod is in conformance with approved rod pattern. In addition, continuation of a reactor startup if the RWM becomes inoperable after the first control rod is withdrawn but before the twelfth control rod is withdrawn will now be controlled limiting this occurrence to once per calendar year. Currently, there is no limit to these occurrences. This change has been previously analyzed and found to be acceptable the NRC in their review of NEDE-24011-P-A, Amendment 17, since the change did not constitute a significant reduction in the margin of safety (the acceptance criterion of $\leq$ 280 cal/gm for the design basis rod drop accident is met even when this change is implemented).

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.3.7.1 - CRAF SYSTEM INSTRUMENTATION

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change allows an extension of the time to place the CRAF subsystem in the pressurization mode of up to 7 days and also allows the same amount of time for one channel inoperable in a trip system as for two channels inoperable in a trip system. The CRAF System instrumentation is not assumed to be an initiator of any analyzed event. The role of the instrumentation is to mitigate and thereby limit the consequences of a design basis accident. The instrumentation actuates to ensure the CRAF System is initiated to ensure main control room dose is limited during a design basis accident. The proposed change to the Actions will not allow continuous operation such that a single failure will preclude CRAF System initiation from mitigating the consequences of a design basis accident. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated. D

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

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   No significant reduction in a margin of safety is involved with this change since the Required Actions either actuate the associated equipment (CRAF subsystem) or require a declaration that the associated equipment is inoperable (which subsequently requires actuation of the CRAF subsustem). Compensatory actions currently exist for this second action for when the CRAF subsystem is inoperable for reasons other than instrumentation inoperabilities. This change also provides a benefit through the potential avoidance of a plant shutdown when alternate compensatory measures are available to ensure the intended function of the instrumentation and associated equipment is satisfied.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Requirements of the LCO not met for reasons other than Condition A, C, D, or F.	G.1 Satisfy the requirements of the LCO.	24 hours
H. Required Action and associated Completion Time of Condition G not met.	H.1 Be in MODE 3.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Verify source of unidentified LEAKAGE increase is not intergranular stress corrosion cracking susceptible material.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations, and RCS inservice leak and hydrostatic testing. ---------- Verify: a. RCS pressure and RCS temperature are within the applicable limits specified in Figures 3.4.11-1, 3.4.11-2, 3.4.11-3 for Unit 1, and Figures 3.4.11-4, 3.4.11-5, and 3.4.11-6 for Unit 2; b. RCS heatup and cooldown rates are $\leq$ 100°F in any 1 hour period; and c. RCS temperature change during system leakage and hydrostatic testing is $\leq$ 20°F in any one hour period when the RCS pressure and RCS temperature are not within the limits of Figure 3.4.11-2 for Unit 1 and Figure 3.4.11-5 for Unit 2.	30 minutes
SR 3.4.11.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.11-3 for Unit 1 and Figure 3.4.11-6 for Unit 2.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

BASES

ACTIONS (continued)

F.1 and G.1

With both recirculation loops operating but the flows not matched, the flows must be matched within 2 hours. If matched flows are not restored, the recirculation loop with lower flow must be declared "not in operation," as required by Required Action F.1. This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing flow control valve position to re-establish forward flow or by tripping the pump.

With the requirements of the LCO not met for reasons other than Conditions A, C, D, and F (e.g., one loop is "not in operation"), compliance with the LCO must be restored within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits for greater than 2 hours (i.e., Required Action F.1 has been taken). Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status. D

Alternatively, if the single loop requirements of the LCO are applied to the APLHGR and MCPR operating limits and RPS and RBM Allowable Values, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 2 hour and 24 hour Completion Times are based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected. D

(continued)

BASES (continued)

ACTIONS

A.1

With RCS unidentified or total LEAKAGE greater than the limits, actions must be taken to reduce the leak. Because the LEAKAGE limits are conservatively below the LEAKAGE that would constitute a critical crack size, 4 hours is allowed to reduce the LEAKAGE rates before the reactor must be shut down. If an unidentified LEAKAGE has been identified and quantified, it may be reclassified and considered as identified LEAKAGE. However, the total LEAKAGE limit would remain unchanged.

B.1 and B.2

An unidentified LEAKAGE increase of > 2 gpm within a 24 hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE increase within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE increase to within limits (i.e., reducing the leakage rate such that the current rate is less than the "2 gpm increase in the previous 24 hours" limit; either by isolating the source or other possible methods) is to verify the source of the unidentified leakage increase is not material susceptible to IGSCC. D

The 4 hour Completion Time is needed to properly reduce the LEAKAGE increase or verify the source before the reactor must be shut down. D

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met or if pressure boundary LEAKAGE exists, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

3/4.4 REACTOR COOLANT SYSTEM [A.1]

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION [A.2]

LCO 3.4.1 3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION

LCO 3.4.1 a. With only one (1) reactor coolant system recirculation loop in operation, comply with Specification 3.4.1.5 and: [within Region III of Figure 3.4.1-1] [A.3]

ACTION G 1. Within four (4) hours: [24] [satisfy the requirements of the LCO] [L.1] [A.2] [D]

a) Place the recirculation flow control system in the Master Manual mode or lower, and [LA.1]

b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 per Specification 2.1.2, and [A.4]

LCO 3.4.1 [as specified in the COLR for Single Loop Operation]

c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation by 0.01 per Specification 3.2.3, and, [LA.6]

d) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable to single recirculation loop operation per Specifications 2.2.1 and 3.3.6. [A.5]

e) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) Limiting Condition for Operation by the applicable Single Loop Operation (SLO) factor specified in the CORE OPERATING LIMITS REPORT.

ACTION H 2. Otherwise, be in at least HOT SHUTDOWN within the next twelve (12) hours.

ACTION D b. With no reactor coolant recirculation loops in operation:

1. Take the ACTION required by Specification 3.4.1.5 and [A.3]

2. Be in at least HOT SHUTDOWN within the next six (6) hours. [12] [L.2]

3/4.4 REACTOR COOLANT SYSTEM

A.1

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

A.2

LIMITING CONDITION FOR OPERATION

LCO 3.4.1 **3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.**

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

A.3

ACTION

LCO 3.4.1 **a. With only one (1) reactor coolant system recirculation loop in operation, ~~comply with Specification 3.4.1.5~~ and:**

within Region III of Figure 3.4.1-1

L.1

ACTION G **1. Within ~~four (4)~~ hours:** 24

satisfy the requirements of the LCO

A.2 D

a) ~~Place the recirculation flow control system in the Master Manual mode or lower, and~~

LA.1

b) ~~Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 per Specification 2.1.2, and~~

A.4

LA.6

as specified in the COLR for Single Loop Operation

c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation ~~by 0.01~~ per Specification 3.2.3, and,

A.5

LCO 3.4.1

d) Reduce the Average Power Range Monitor (APRM) Scram ~~and Rod Block~~ and Rod Block Monitor ~~Trip Setpoints~~ and Allowable Values to those applicable to single recirculation loop operation per Specifications 2.2.1 and 3.3.6.

e) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) Limiting Condition for Operation by the applicable Single Loop Operation (SLO) factor specified in the CORE OPERATING LIMITS REPORT.

ACTION H **2. Otherwise, be in at least HOT SHUTDOWN within the next twelve (12) hours.**

ACTION D **b. With no reactor coolant recirculation loops in operation:**

~~1. Take the ACTION required by Specification 3.4.1.5, and~~

A.3

2. Be in at least HOT SHUTDOWN within ~~the next six (6)~~ hours.

12

L.2

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 The time to adjust power distribution limits and Reactor Protection System and Control Rod Block instrumentation Allowable Values for single loop operation in CTS 3.4.1.1 Action a.1 is proposed to be increased from 4 hours to 24 hours (proposed Required Action G.1 Completion Time). The increased Completion Time to perform the power distribution limits and instrument adjustments for single recirculation loop operation, considering the time required to secure the necessary resources (e.g., notifying appropriate personnel, obtaining equipment needed to perform the adjustments, and performing appropriate per-job briefing for the RPS and Control Rod Block instrumentation adjustments), is reasonable to avoid unnecessary transients on the plant. The 24 hour Completion Time to adjust the power distribution limits and instrumentation for single recirculation loop operation is considered acceptable based on the low probability of an accident occurring during this period and frequent core monitoring by operations allowing abrupt changes in core flow conditions to be quickly detected. These proposed changes are offset by the benefit of not hastily adjusting the instrumentation for single loop operation which could increase the probability of a plant transient. D D

L.2 In the event no recirculation loops are in operation, the time required to shutdown in ITS 3.4.1 (Required Action D.3) is 12 hours versus the 6 hour time period allowed by CTS 3.4.1.1 Action b.2 and CTS 3.4.1.5 Action a.2.c). In this degraded condition with no recirculation loops in operation, a Completion Time of 12 hours to be in MODE 3 (Hot Shutdown) provides a reasonable time period to place the unit in a MODE in which the LCO does not apply and is consistent with the Completion Time of similar Technical Specification required plant shutdowns. This change is considered acceptable since in a natural circulation condition, the severity of a DBA is reduced, and there is minimal dependence on the recirculation loop coastdown characteristics. Allowing 12 hours to reach MODE 3 is an acceptable exchange in risk; the risk of a DBA or instability during the additional period to reach MODE 3, versus the potential risk of a unit upset that could challenge safety systems resulting from a rapid plant shutdown.

L.3 CTS 4.4.1.3 requires the recirculation loop flow mismatch to be verified within the limits once per 24 hours when in Operational Condition 1 and 2 during two recirculation loop operation. CTS 4.0.4 requires the Surveillances to be met prior to entry into the applicable Mode or other specified conditions.
CTS 4.4.1.3 cannot be performed prior to its Applicability if shifting from single loop to two loop operation while in MODE 1 or 2. Therefore, a note has been

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

<LCO 3.4.1.1> LCO 3.4.1 Two recirculation loops with matched flows shall be in operation, within Region III of Figure 3.4.1-1 [1]

OR

One recirculation loop ~~may~~ shall be in operation ~~provided~~ with the following limits ~~are~~ applied when the associated LCO is applicable: [1]

<3.4.1 Act a
3.4.1 Act a.1.c)
3.4.1 Act a.1.d)
3.4.1 Act a.1.e)>

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR; [2]

<LCO 3.4.1.3>

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR; ~~and~~ [1]

<LCO 3.4.1.5
LCO 3.4.1.5.b>

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power—~~High~~ Upscale), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation; ~~.~~ and [1] [2] [3]

d. LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 1.a (Rod Block Monitor—Upscale), Allowable Value of Table 3.3.2.1-1, specified in the COLR, is reset for single loop operation. [1]

<Appl 3.4.1.1
Appl 3.4.1.3
Appl 3.4.1.5>

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A.~~ G [1] Requirements of the LCO not met.	~~A.~~ G 1 [1] Satisfy the requirements of the LCO.	24 hours

<3.4.1.1 Act a.1>

~~* Pending resolution of stability issue.~~ [1]

(continued)

For reasons other than Condition A, C, D, or F [1]

Insert 3.4.1-ACTIONS [1]

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.3.2 Act e> B. (continued) is not intergranular stress corrosion cracking susceptible material.	B.2 Verify source of unidentified LEAKAGE increase ~~is not service sensitive type 304 or type 316 austenitic stainless steel.~~	4 hours [4]
<3.4.3.2 Act a 3.4.3.2 Act b 3.4.3.2 Act e> C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.4.3.2.1> SR 3.4.5.1	Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	8 hours 12 [5]

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.6.1 Act> <DOC M.2> C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. ---------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
	C.2 Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.11.1 ----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations, and RCS inservice leak and hydrostatic testing. ---------- <LCO 3.4.6.1> <LCO 3.4.6.1.a> <LCO 3.4.6.1.b> <LCO 3.4.6.1.c> <4.4.6.1> a. Verify RCS pressure and RCS temperature ~~and RCS heatup and cooldown rates~~ are within the applicable limits specified in ~~the PTLR.~~	30 minutes
<LCO 3.4.6.1> <4.4.6.1.2> SR 3.4.11.2 Verify RCS pressure and RCS temperature are within the criticality limits specified in ~~the PTLR.~~ Figure 3.4.11-3 for Unit 1 and Figure 3.4.11-6 for Unit 2	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

Figures 3.4.11-1, 3.4.11-2, and 3.4.11-3 for Unit 1, and 3.4.11-4, 3.4.11-5, and 3.4.11-6 for Unit 2;

b. RCS heatup and cooldown rates are ≤ 100°F in any 1 hour period; and

c. RCS temperature change during system leakage and hydrostatic testing is ≤ 20°F in any one hour period when the RCS pressure and RCS temperature are not within the limits of Figure 3.4.11-2 for Unit 1 and Figure 3.4.11-5 for Unit 2.

1 D D D

BASES

LCO (continued)

applied to allow continued operation consistent with the assumptions of Reference 3. <Insert LCO> 1

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor ~~Coolant~~ 2 Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important. 4

ACTIONS

~~A.1~~ F.1 and G.1 4

4 <Insert ACTIONS>

With both recirculation loops operating but the flows not matched, the flows must be matched within 2 hours. If matched flows are not restored, the recirculation loop with lower flow must be declared "not in operation" as required by Required Action F.1.

~~With the requirements of the LCO not met, the recirculation loops~~ must be restored ~~to operation with matched flows~~ within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. ~~The loop with the lower flow must be considered not in operation.~~ Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS ~~setpoints~~ and RBM Allowable Values, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence. 4

The 24 hour Completion ~~Time is~~ Times are based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large 4

(continued)

BASES (continued)

ACTIONS

A.1

With RCS unidentified or total LEAKAGE greater than the limits, actions must be taken to reduce the leak. Because the LEAKAGE limits are conservatively below the LEAKAGE that would constitute a critical crack size, 4 hours is allowed to reduce the LEAKAGE rates before the reactor must be shut down. If an unidentified LEAKAGE has been identified and quantified, it may be reclassified and considered as identified LEAKAGE. However, the total LEAKAGE limit would remain unchanged.

B.1 and B.2

An unidentified LEAKAGE increase of > 2 gpm within a 4 hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE increase within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE increase to within limits (i.e., reducing the leakage rate such that the current rate is less than the "2 gpm increase in the previous 4 hours" limit; either by isolating the source or other possible methods) is to evaluate RCS type 304 and type 316 austenitic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids and determine it is not the source of the increased LEAKAGE. This type of piping is very susceptible to IGSCC.

verify the source of the unidentified leakage increase is not material susceptible to IGSCC.

The 4 hour Completion Time is needed to properly reduce the LEAKAGE increase or verify the source before the reactor must be shut down.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met or if pressure boundary LEAKAGE exists, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable,

(continued)

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.1 - RECIRCULATION LOOPS OPERATING

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change increases the time to adjust power distribution limits and Reactor Protection System (RPS) and Control Rod Block instrumentation Allowable Values for single loop operation to 24 hours. The time required to perform power distribution limit adjustments or RPS or Control Rod Block instrumentation adjustment is not considered as an initiator of any accidents previously evaluated. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Additionally, the consequences of an event occurring while the unit is in single loop operation without the power distribution limit adjustments or RPS or Control Rod Block instrumentation adjustments are not altered by the proposed change. Therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

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

   The proposed change does not involve new equipment, design or operations, but provides for additional time to complete the previously approved TS Actions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change allows additional time to adjust power distribution limits and Reactor Protection System (RPS) and Control Rod Block instrumentation Allowable Values. However, this condition occurs infrequently and any minor decrease in the margin during this additional time is offset by the benefit of not hastily adjusting the instrumentation for single loop operation which could increase the probability of a plant transient. The additional time to adjust the power distribution limits and instrumentation for single recirculation loop operation is considered acceptable based on the low probability of an accident occurring during this period and frequent core monitoring by operations allowing abrupt changes in core flow conditions to be quickly detected. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

BASES

APPLICABLE SAFETY ANALYSES (continued)

capability to adequately cool the core, under near-term and long-term conditions, and prevent excessive fuel damage. For all LOCA analyses, only six ADS valves are assumed to function. An additional analysis has been performed which assumes five ADS valves function, however in this analysis all low pressure and high pressure ECCS subsystems are also assumed to be available.

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

LCO

Each ECCS injection/spray subsystem and six ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the three LPCI subsystems, the LPCS System, and the HPCS System. The low pressure ECCS injection/spray subsystems are defined as the LPCS System and the three LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in 10 CFR 50.46 (Ref. 10) could potentially be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by 10 CFR 50.46 (Ref. 10).

As noted, LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR cut in permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes: a) when the system is realigned to or from the RHR shutdown cooling mode and; b) when the system is in the RHR shutdown cooling mode, whether or not the RHR pump is operating. This allowance is necessary since the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.

APPLICABILITY

All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3 when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system

(continued)

BASES

APPLICABILITY (continued)

piping. In MODES 2 and 3, the ADS function is not required when pressure is ≤ 150 psig because the low pressure ECCS subsystems (LPCS and LPCI) are capable of providing flow into the RPV below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS—Shutdown.

ACTIONS

A.1

If any one low pressure ECCS injection/spray subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE subsystems concurrent with a LOCA may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is based on a reliability study (Ref. 12) that evaluated the impact on ECCS availability by assuming that various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (i.e., Completion Times).

B.1 and B.2

If the HPCS System is inoperable, and the RCIC System is immediately verified to be OPERABLE (when RCIC is required to be OPERABLE), the HPCS System must be restored to OPERABLE status within 14 days. In this Condition, adequate core cooling is ensured by the OPERABILITY of the redundant and diverse low pressure ECCS injection/spray subsystems in conjunction with the ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Immediate verification of RCIC OPERABILITY is therefore required when HPCS is inoperable and RCIC is required to be OPERABLE. This may be performed by an administrative check, by examining logs or other information, to determine if RCIC is out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the RCIC System. However, if the OPERABILITY

(continued)

BASES

LCO (continued)

local) to the LPCI mode and is not otherwise inoperable. Alignment and operation for decay heat removal includes: a) when the system is realigned to or from the RHR shutdown cooling mode and; b) when the system is in the RHR shutdown cooling mode, whether or not the RHR pump is operating. This allowance is necessary since the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Because of low pressure and low temperature conditions in MODES 4 and 5, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncovery.

APPLICABILITY

OPERABILITY of the ECCS injection/spray subsystems is required in MODES 4 and 5 to ensure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel in the core in case of an inadvertent draindown of the vessel. Requirements for ECCS OPERABILITY during MODES 1, 2, and 3 are discussed in the Applicability section of the Bases for LCO 3.5.1. ECCS subsystems are not required to be OPERABLE during MODE 5 with the spent fuel storage pool gates removed and the water level maintained at $\geq$ 22 ft above the RPV flange. This provides sufficient coolant inventory to allow operator action to terminate the inventory loss prior to fuel uncovery in case of an inadvertent draindown.

The Automatic Depressurization System is not required to be OPERABLE during MODES 4 and 5 because the RPV pressure is < 150 psig, and the LPCS, HPCS, and LPCI subsystems can provide core cooling without any depressurization of the primary system.

ACTIONS

A.1 and B.1

If any one required ECCS injection/spray subsystem is inoperable, the required inoperable ECCS injection/spray subsystem must be restored to OPERABLE status within 4 hours. In this Condition, the remaining OPERABLE subsystem can provide sufficient RPV flooding capability to recover from an inadvertent vessel draindown. However, overall system reliability is reduced because a single failure in the remaining OPERABLE subsystem concurrent with

(continued)

BASES

LCO (continued)

injection/spray subsystems are defined as the LPCS System and the three LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in 10 CFR 50.46 (Ref. 10) could potentially be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by 10 CFR 50.46 (Ref. 10).

As noted, LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR cut in permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Insert LCO At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.

APPLICABILITY

All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3 when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, the ADS function is not required when pressure is ≤ 150 psig because the low pressure ECCS subsystems (LPCS and LPCI) are capable of providing flow into the RPV below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS—Shutdown."

ACTIONS

A.1

If any one low pressure ECCS injection/spray subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE subsystems concurrent with a LOCA may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is based on a reliability study (Ref. 12) that evaluated the impact on ECCS availability by assuming that various components and subsystems were taken

(continued)

Insert LCO

Alignment and operation for decay heat removal includes: a) when the system is realigned to or from the RHR shutdown cooling mode and; b) when the system is in the RHR shutdown cooling mode, whether or not the RHR pump is operating. This allowance is necessary since the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor.

INSERT B 3.5-14(A)

Alignment and operation for decay heat removal includes: a) when the system is realigned to or from the RHR shutdown cooling mode and; b) when the system is in the RHR shutdown cooling mode, whether or not the RHR pump is operating. This allowance is necessary since the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor.

D

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.1.2	Verify primary containment structural integrity in accordance with the Inservice Inspection Program for Post Tensioning Tendons.	In accordance with the Inservice Inspection Program for Post Tensioning Tendons
SR 3.6.1.1.3	Verify drywell-to-suppression chamber bypass leakage is $\leq$ 10% of the acceptable $A/\sqrt{k}$ design value of 0.030 ft^2 at an initial differential pressure of $\geq$ 1.5 psid.	24 months AND ------NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass ---------- 12 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is ≥ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.3	Verify the secondary containment can be drawn down to ≥ 0.25 inch of vacuum water gauge in ≤ 300 seconds using one standby gas treatment (SGT) subsystem.	24 months on a STAGGERED TEST BASIS for each SGT subsystem
SR 3.6.4.1.4	Verify the secondary containment can be maintained ≥ 0.25 inch of vacuum water guage for 1 hour using one SGT subsystem at a flow rate ≤ 4400 cfm.	24 months on a STAGGERED TEST BASIS for each SGT subsystem
SR 3.6.4.1.5	Verify all secondary containment equipment hatches are closed and sealed.	24 months

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.1.3

Maintaining the pressure suppression function of the primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR measures drywell-to-suppression chamber differential pressure during a 1 hour period to ensure that the leakage paths that would bypass the suppression pool are within allowable limits.

Satisfactory performance of this SR can be achieved by establishing a known differential pressure ($\geq$ 1.5 psid) between the drywell and the suppression chamber and verifying that the measured bypass leakage is $\leq$ 10% of the acceptable $A/\sqrt{k}$ design value of 0.030 ft^2. The leakage test is performed every 24 months. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed during a unit outage and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation, in this event, as the Note indicates, increasing the Frequency to once every 12 months is required until the situation is remedied as evidenced by passing two consecutive tests.

REFERENCES

1. UFSAR, Section 6.2.

2. UFSAR, Section 15.6.5.

3. 10 CFR 50, Appendix J, Option B.

4. UFSAR, Section 6.2.6.1.

5. Regulatory Guide 1.35, Revision 3.

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

Required Action C.1 has 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. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

This SR ensures that the secondary containment boundary is sufficiently leak tight to preclude exfiltration. The 24 hour Frequency of this SR was developed based on operating experience related to secondary containment vacuum variations during the applicable MODES and the low probability of a DBA occurring.

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal secondary containment vacuum condition.

SR 3.6.4.1.2 and SR 3.6.4.1.5

Verifying that one secondary containment access door in each access opening is closed and each equipment hatch is closed and sealed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. In addition, for equipment hatches that are floor plugs, the "sealed" requirement is effectively met by gravity. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. An access opening contains one inner and one outer door. In some cases a secondary containment barrier contains multiple inner or multiple outer doors. For these cases, the access openings share the inner door or the outer door, i.e., the access openings have

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.2 and SR 3.6.4.1.5 (continued) D

a common inner door or outer door. The intent is to not breach the secondary containment at any time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times, i.e., all inner doors closed or all outer doors closed. Thus each access opening has one door closed. However, each secondary containment access door is normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on the access opening. The 31 day Frequency for SR 3.6.4.1.2 has been shown to be adequate based on operating experience, and is considered adequate in view of the existing administrative controls on door status. The 24 month Frequency for SR 3.6.4.1.5 is considered adequate in view of the existing administrative controls on equipment hatches. D D

SR 3.6.4.1.3 and SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. Each SGT subsystem is designed to drawdown pressure in the secondary containment to $\geq$ 0.25 inches of vacuum water gauge in $\leq$ 300 seconds and maintain pressure in the secondary containment at $\geq$ 0.25 inches of vacuum water gauge for 1 hour at a flow rate of $\leq$ 4400 cfm. To ensure that all fission products released to secondary containment are treated, SR 3.6.4.1.3 and SR 3.6.4.1.4 verify that a pressure in the secondary containment that is less than the pressure external to the secondary containment boundary can rapidly be established and maintained. When the SGT System is operating as designed, the establishment and maintenance of secondary containment pressure cannot be accomplished if the secondary containment boundary is not intact. Establishment of this pressure is confirmed by SR 3.6.4.1.3, which demonstrates that the secondary containment can be drawn down to $\geq$ 0.25 inches of vacuum water gauge in $\leq$ 300 seconds using one SGT subsystem. SR 3.6.4.1.4 demonstrates that the pressure in the secondary containment can be maintained $\geq$ 0.25 inches of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate $\leq$ 4400 cfm. This flow rate is the assumed secondary containment leak rate during the drawdown period. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady

(continued)

SUPPRESSION CHAMBER# A.1 ITS 3.6.1.1

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a level between +3 inches** and -4 1/2 inches**, and a

2. Maximum average temperature of 105°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

b) 120°F with the main steam line isolation valves closed following a scram.

LCO 3.6.1.1 b. Drywell-to-suppression charmer bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.03 ft^2. A.6 D

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 105°F, stop all testing which adds heat to the suppression pool, and restore the average temperature to less than or equal to 105°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

2. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.
**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

<See ITS 3.6.2.1 and ITS 3.6.2.2>

A.1

ITS 3.6.1.1

SURVEILLANCE REQUIREMENTS (Continued)

c. Deleted. LD.1

SR 3.6.1.1.3

d. By conducting drywell-to-suppression chamber bypass leak tests at least once per 18 months at an initial differential pressure of 1.5 psi and verifying that the A/√k calculated from the measured leakage is within the specified limit. A.6 D

If any 1.5 psi leak test results in a calculated A/√k >20% of the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission. L5

SR 3.6.1.1.3

If two consecutive 1.5 psi leak tests result in a calculated A/√k greater than the specified limit, then:

1. A 1.5 psi leak test shall be performed at least once per 9 months until two consecutive 1.5 psi leak tests result in the calculated A/√k within the specified limits, and LD.1 C

2. A 5 psi leak test, performed with the second consecutive successful 1.5 psi leak test, results in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

If any required 5 psi leak test results in a calculated A/√k greater than the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission. L2

If two consecutive 5 psi leak tests result in a calculated A/√k greater than the specified limit, then a 5 psi leak test shall be performed at least once per 9 months until two consecutive 5 psi leak tests result in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER#

A.1

ITS 3.6.1.1

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a level between +3 inches** and -4 1/2 inches**, and a

2. Maximum average temperature of 105°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

b) 120°F with the main steam line isolation valves closed following a scram.

LCO 3.6.1.1

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.03 ft^2.

A.6

D

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 105°F, stop all testing which adds heat to the suppression pool, and restore the average temperature to less than or equal to 105°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

2. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.
**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

<See ITS 3.6.2.1 and ITS 3.6.2.2>

A.1

ITS 3.6.1.1

SURVEILLANCE REQUIREMENTS (Continued)

c. Deleted. LD.1

SR 3.6.1.1.3

d. By conducting drywell-to-suppression chamber bypass leak tests at least once per 24 months at an initial differential pressure of 1.5 psi and verifying that the A/√k calculated from the measured leakage is within the specified limit. A.6 D

~~If any 1.5 psi leak test results in a calculated A/√k >20% of the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.~~ L.5

If two consecutive 1.5 psi leak tests result in a calculated A/√k greater than the specified limit, then:

1. A 1.5 psi leak test shall be performed at least once per 12 months until two consecutive 1.5 psi leak tests result in the calculated A/√k within the specified limits, and LD.1 C

2. ~~A 5 psi leak test, performed with the second consecutive successful 1.5 psi leak test, results in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.~~

~~If any required 5 psi leak test results in a calculated A/√k greater than the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.~~ L.2

~~If two consecutive 5 psi leak tests result in a calculated A/√k greater than the specified limit, then a 5 psi leak test shall be performed at least once per 9 months until two consecutive 5 psi leak tests result in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.~~

ADMINISTRATIVE

A.6 (cont'd) for the primary containment to perform its pressure suppression function and to ensure the primary containment design pressure is not exceeded. Therefore, the actual LCO statement is not needed since it is part of Primary Containment OPERABILITY (ITS 3.6.1.1). This change is considered a presentation preference, which is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Not used. D

LD.1 The Frequency for performing CTS 4.6.2.1.d (proposed SR 3.6.1.1.3), the drywell-to-suppression chamber bypass leak test, has been extended from 18 months to 24 months for the routine test and from 9 months to 12 months for additional tests (1.5 psi leak test) required if a routine test fails two times in a row, to facilitate a change to the LaSalle 1 and 2 refuel cycle from 18 months to 24 months. The proposed change will allow the normal Surveillance to extend the Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed Specification 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. C

SR 3.6.1.1.3 verifies the drywell-to-suppression chamber bypass leakage is less than or equal to the bypass leakage limit. The leakage test is performed every 24 months, consistent with the requirement to perform the test during a refueling outage, risk of high radiation exposure, and the remote possibility of a component failure that is not identified by other drywell or primary containment SR.

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis. Since the current 9 month Frequency is based on reducing the normal 18 month Frequency by half (performing CTS 4.6.2.1.d twice as often), it has been changed to 12 months (half the proposed 24 month normal Frequency). C

"Specific"

L.1 In the ITS presentation (refer to Discussion of Change A.6 above), drywell-to-suppression chamber bypass leakage outside limits (proposed SR 3.6.1.1.3) will result in declaring the Primary Containment inoperable. ITS 3.6.1.1 ACTIONS for these conditions require commencing a shutdown to MODES 3 and 4 if the leakage problem is not corrected within 1 hour. CTS 3.6.2.1 Action e only restricts heating up reactor coolant above 200°F (i.e., entry into MODE 3). With the drywell-to-suppression chamber bypass leakage outside of limits in MODE 1, 2, or 3, CTS 3.6.2.1 does not provide actions. Since drywell-to-suppression chamber leakage are attributes of maintaining Primary Containment Integrity (in ITS terminology, primary containment OPERABILITY), a 1 hour allowed outage time is provided for this condition consistent with the primary containment is inoperable. This change will provide consistency in ITS ACTIONS for the various primary containment degradations. With primary containment OPERABILITY lost, the risk associated with continued operation for a short period of time could be less than that associated with an immediate plant shutdown. This change to CTS 3.6.2.1 is acceptable due to the low probability of an event that could pressurize the primary containment during the short time in which continued operation is allowed and primary containment is inoperable.

L.2 The accelerated test basis and elevated test pressure requirements of CTS 4.6.2.1.d.2 are deleted. CTS 4.6.2.1.d.2 requires verification of drywell-to-suppression chamber bypass leakage on an accelerated test basis and at a higher test pressure in the event that the results of consecutive drywell-to-suppression chamber bypass leakage tests are outside Technical Specification specified limits. Under the proposed change, drywell-to-suppression chamber will continue to be verified on the frequency and at the test pressure described in CTS 4.6.2.1.d. Performance of drywell-to-suppression chamber on an accelerated test basis and

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 (cont'd) at elevated test pressure is not considered to be advantageous for LaSalle 1 and 2 based upon the satisfactory results obtained from previous drywell-to-suppression pool leakage tests. The original, as issued, Unit 1 and 2 LaSalle Technical Specifications included a requirement to perform a drywell-to-suppression chamber bypass leakage test at the elevated pressure during the first refueling outage and then on the schedule required for Type A overall integrated containment leakage rate tests by CTS 4.6.1.2.a. The elevated pressure test was allowed to be discontinued if the first two elevated pressure tests resulted in the calculated $A/\sqrt{k}$ being within the specified limit and the $A/\sqrt{k}$ calculated from the 1.5 psi leak tests was within 20% of the specified limit. This comparison of tests results between the 1.5 psi test and the elevated pressure test confirmed that the 1.5 psi test was adequate to verify bypass leakage is within the specified limits. These requirements were deleted in Amendments 102 and 87 for Units 1 and 2, respectively, since, as stated in the NRC SER for these amendments (letter from W. D. Reckley to D. L. Farrar, dated March 16, 1995), the criteria for discontinuing the elevated pressure tests were satisfied. Additionally, the acceptance criteria for drywell-to-suppression chamber bypass leakage measured during testing is small compared to the drywell-to-suppression chamber leakage assumed in the accident analyses, and is limited to 10% of the design value specified in the UFSAR. Consequently, the change is acceptable because it has no adverse impact on primary containment structural integrity or plant operations. D

L.3 Not used. D

L.4 Not used. C

L.5 The requirement in CTS 4.6.2.1.d for the NRC to review the test schedule for subsequent tests if any leak rate test result is not within the required limits has been deleted since the NRC has already approved the test schedule. If one test fails, the current Technical Specifications do not require the test frequency to be changed. The test frequency is only required to be changed if two consecutive tests have failed, as stated in CTS 4.6.2.1.d. Since the test schedule is already covered by the Technical Specifications, which has been approved by the NRC, there is no reason to have a requirement that the NRC review the test schedule (which will not change from the current test schedule) when one test fails. In addition, a historical review has shown this Surveillance has never failed. Therefore, this change is considered to be acceptable.

RELOCATED SPECIFICATIONS

None

A.1

ITS 3.6.4.1

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

LCO 3.6.4.1

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained. OPERABLE A.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY: OPERABLE

ACTION A — a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION B

ACTION C — b. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY OPERABILITY shall be demonstrated by: A.2

SR 3.6.4.1.1 a. Verifying at least once per 24 hours that the pressure within the secondary containment is less than or equal to 0.25 inches of vacuum water gauge.# M.1

b. Verifying at least once per 31 days that:

SR 3.6.4.1.2 1. At least one door in each access to the secondary containment is closed. A.3

2. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position. A.4 moved to ITS 3.6.4.2

c. At least once per 18 24 months; LD.1 on a STAGGERED TEST BASIS M.2

SR 3.6.4.1.3 1. Verifying that one standby gas treatment subsystem will draw down the secondary containment to greater than or equal to 0.25 in. of vacuum water gauge in less than or equal to 300 seconds, and

SR 3.6.4.1.4 2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment at a flow rate not exceeding 4000 CFM ± 10%.

add proposed SR 3.6.4.1.5 M.3 D

Applicability

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#SECONDARY CONTAINMENT INTEGRITY is maintained when secondary containment vacuum is less than required for up to 1 hour solely due to Reactor Building ventilation system failure. M.1

ITS 3.6.4.1

A.1

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

LCO 3.6.4.1 3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be ~~maintained~~ OPERABLE — A.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY: OPERABLE

ACTION A — a. In OPERATIONAL CONDITION 1, 2, or 3, restore SECONDARY CONTAINMENT INTEGRITY to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION B

ACTION C — b. In OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS — A.2

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY OPERABILITY shall be demonstrated by:

SR 3.6.4.1.1 a. Verifying at least once per 24 hours that the pressure within the secondary containment is less than or equal to 0.25 inch of vacuum water gauge. — M.1

b. Verifying at least once per 31 days that:

SR 3.6.4.1.2 1. At least one door in each access to the secondary containment is closed. — A.3

2. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position. — A.4 moved to ITS 3.6.4.2 — LD.1

c. At least once per 18 24 months; ON a STAGGERED TEST BASIS — M.2

SR 3.6.4.1.3 1. Verifying that one standby gas treatment subsystem will draw down the secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 300 seconds, and

SR 3.6.4.1.4 2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the secondary containment at a flow rate not exceeding 4000 cfm ± 10%.

add proposed SR 3.6.4.1.5 — M.3 — D

Applicability

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#SECONDARY CONTAINMENT INTEGRITY is maintained when secondary containment vacuum is less than required for up to 1 hour solely due to Reactor Building ventilation system failure. — M.1

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 (cont'd) deleted. The existing and proposed 4 hour Completion Time in CTS 3.6.5.1 Action A and ITS 3.6.4.1 ACTION A, respectively, provides adequate time to re-establish secondary containment vacuum. If the secondary containment vacuum cannot be maintained due to loss of the Reactor Building Ventilation System, then the secondary containment is inoperable and the ACTIONS should be entered immediately, not delayed for an hour, consistent with the loss of secondary containment for any other reason. This is an additional restriction on plant operation.

M.2 CTS 4.6.5.1.c requires that one subsystem be tested every 18 months. However, the same SGT subsystem could be tested at each testing occurrence. Proposed SR 3.6.4.1.3 and SR 3.6.4.1.4 will now require both subsystems be tested in the course of 48 months, as represented by the Staggered Test Basis requirement of the 24 month Frequency. This will ensure each SGT subsystem can maintain the proper vacuum. This is an additional restriction on plant operation.

M.3 A new Surveillance is being added, ITS SR 3.6.4.1.5, which requires all secondary containment equipment hatches to be verified closed and sealed every 24 months. This SR provides adequate assurance that exfiltration from the secondary containment through these hatches will not occur. D

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LD.1 The Frequencies for performing CTS 4.6.5.1.c.1 and 4.6.5.1.c.2 have been extended from 18 months to 24 months in proposed SR 3.6.4.1.3 and SR 3.6.4.1.4 to facilitate a change to the LaSalle 1 and 2 refuel cycle from 18 months to 24 months. These surveillances ensure that the Secondary Containment is OPERABLE to support the drawdown analysis. The proposed change will allow these Surveillances to extend the Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed Specification 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) SR 3.6.4.1.3 verifies the secondary containment can be drawn down to the specified vacuum in the time required using one standby gas treatment (SGT) subsystem. SR 3.6.4.1.4 verifies the secondary containment can be maintained at the specified vacuum for the required time using one SGT subsystem at the specified flow rate. The purpose of these tests is to ensure secondary containment boundary integrity by demonstrating that secondary containment vacuum assumed in the safety analysis can be established and maintained under design basis conditions. Extending the Surveillance interval for this verification of secondary containment integrity is acceptable because secondary containment is maintained at a negative pressure during normal operation, and secondary containment structural integrity is maintained through administrative controls which ensure that no significant changes will be made to the secondary containment structure without proper evaluation. Furthermore, based on engineering judgement, any structural degradation which would result in impacting secondary containment OPERABILITY is not likely to occur during normal plant operation. Any event which would cause significant structural degradation, such as a seismic event, would require a plant evaluation.

Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. In addition, the proposed 24 month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

"Specific"

None

RELOCATED SPECIFICATIONS

None

⟨CTS⟩

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
⟨4.6.1.1.d⟩	SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~ the Primary Containment Leakage Rate Testing Program ~~The leakage rate acceptance criterion is ≤ 1.0 L_a. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are < 0.6 L_a for the Type B and Type C tests, and < 0.75 L_a for the Type A test.~~	~~-----NOTE----- SR 3.0.2 is not applicable ----------~~ In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~
⟨4.6.1.1.e⟩	SR 3.6.1.1.2	Verify primary containment structural integrity in accordance with the ~~Primary Containment Tendon Surveillance Program.~~ Inservice Inspection Program for Post Tensioning Tendons	In accordance with the ~~Primary Containment Tendon Surveillance Program~~
	SR 3.6.1.1.3	Verify drywell-to-suppression chamber bypass leakage is ≤ 10% of the acceptable $A\sqrt{K}$ design value of 0.030 ft^2 at an initial differential pressure of ≥ 1.5 psid.	24 months AND --- NOTE --- Only required after two consecutive tests fail and continues until two consecutive tests pass 12 months

1 2 3 D C

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.5.1 Act b> C. Secondary containment inoperable during movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Suspend movement of irradiated fuel assemblies in the primary and secondary containment.	Immediately
	AND	
	C.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.6.5.1.a> SR 3.6.4.1.1	Verify secondary containment vacuum is ≥ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed.	31 days 24 months

(move to after SR 3.6.4.1.4)

(continued)

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.6.4.1 - SECONDARY CONTAINMENT

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. ISTS SR 3.6.4.1.2 requires verification that all secondary containment equipment hatches are closed and sealed every 31 days. This Surveillance Requirement is not required in the LaSalle 1 and 2 CTS. However, the SR will be added at a 24 month Frequency (ITS SR 3.6.4.1.5). At LaSalle 1 and 2, all equipment access openings are provided with inner and outer doors and are treated as access doors. As a result, they will be subject to the verification requirements of ITS SR 3.6.4.1.2 (ISTS SR 3.6.4.1.3), which verifies the position of secondary containment access doors. In addition, the following SRs have been renumbered due to this change.

3. The ISTS SR 3.6.4.1.3 (ITS SR 3.6.4.1.2) allowance that both doors can be open during entry and exit has been deleted. This is consistent with the same SR in NUREG-1433, Rev. 1. The LaSalle 1 and 2 design with respect to the number of doors in an access opening is consistent with the BWR/4 design (2 doors per access opening), not the BWR/6 design (one door per access opening).

4. ISTS SRs 3.6.4.1.4 and 3.6.1.4.5 are tests that ensure the Secondary Containment is Operable; the leak tightness of the Secondary Containment boundary is within the assumptions of the accident analyses. However, they are written in such a manner that they imply that if a SGT subsystem is inoperable, the SRs are failed ("Verify each standby gas treatment (SGT) subsystem will/can..."). As stated above, this is not the intent of the SRs. Therefore, to ensure this misinterpretation cannot occur, the SRs have been rephrased to more clearly convey the original intent of the SRs, to verify the Secondary Containment is Operable. With the new wording, if a SGT subsystem is inoperable, ITS SRs 3.6.4.1.3 and 3.6.4.1.4 will still be met and only the SGT System Specification, LCO 3.6.4.3, will be required to be entered. The SRs will still ensure each SGT subsystem is used (on a STAGGERED TEST BASIS) to perform the SRs. This change is also consistent with TSTF-322.

3 Insert SR 3.6.1.1.3

SR 3.6.1.1.3

Maintaining the pressure suppression function of the primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR measures drywell-to-suppression chamber differential pressure during a 1 hour period to ensure that the leakage paths that would bypass the suppression pool are within allowable limits. D

Satisfactory performance of this SR can be achieved by establishing a known differential pressure ($\geq$ 1.5 psid) between the drywell and the suppression chamber and verifying that the measured bypass leakage is $\leq$ 10% of the acceptable $A/\sqrt{k}$ design value of 0.030 ft^2. The leakage test is performed every 24 months. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed during a unit outage and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation, in this event, as the Note indicates, increasing the Frequency to once every 12 months is required until the situation is remedied as evidenced by passing two consecutive tests. C

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

This SR ensures that the {secondary containment} boundary is sufficiently leak tight to preclude exfiltration ~~under expected wind conditions~~. The 24 hour Frequency of this SR was developed based on operating experience related to {secondary containment} vacuum variations during the applicable MODES and the low probability of a DBA occurring ~~between surveillances~~.

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal {secondary containment} vacuum condition.

SR 3.6.4.1.2 and SR 3.6.4.1.3

Verifying that {secondary containment} equipment hatches and access doors are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the {secondary containment} will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining {secondary containment} OPERABILITY requires verifying each door in the access opening is closed, except when the access opening is being used for entry and exit; then, at least one door must remain closed. The 31 day Frequency for these SRs has been shown to be adequate based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

INSERT B3.6.4.1.2

existing administrative controls on door status. The 24 Month Frequency for SR 3.6.4.1.5 is considered adequate in view of the existing administrative controls on equipment hatches.

In addition, for equipment hatches that are floor plugs, the "sealed" requirement is effectively met by gravity.

(continued)

L.3 CHANGE

Not used. D

3.7 PLANT SYSTEMS

3.7.2 Diesel Generator Cooling Water (DGCW) System

LCO 3.7.2 The following DGCW subsystems shall be OPERABLE:

a. Three DGCW subsystems; and

b. The opposite unit Division 2 DGCW subsystem.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each DGCW subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGCW subsystems inoperable.	A.1 Declare supported component(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify each DGCW subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.2	Verify each DGCW pump starts automatically on each required actual or simulated initiation signal.	24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify cooling water temperature supplied to the plant from the CSCS pond is ≤ 100°F.	24 hours
SR 3.7.3.2	Verify sediment level is ≤ 1.5 ft in the intake flume and the CSCS pond.	24 months
SR 3.7.3.3	Verify CSCS pond bottom elevation is ≤ 686.5 ft.	24 months

B 3.7 PLANT SYSTEMS

B 3.7.2 Diesel Generator Cooling Water (DGCW) System

BASES

BACKGROUND

The DGCW System is designed to provide cooling water for the removal of heat from the standby diesel generators, low pressure core spray (LPCS) pump motor cooling coils, and Emergency Core Cooling System (ECCS) cubicle area cooling coils that support equipment required for a safe reactor shutdown following a design basis accident (DBA) or transient.

The DGCW System consists of three independent cooling water headers (Divisions 1, 2, and 3), and their associated pumps, valves, and instrumentation. The pump and header for the Division 1 DGCW subsystem is common to both units (and supplies cooling to equipment on both units). The other divisions have independent pumps and suction headers.

The following combinations of DGCW pumps are sized to provide sufficient cooling capacity to support the required safety related systems during safe shutdown of the unit following a loss of coolant accident (LOCA):

a. The Division 1 and 2 DGCW pumps; D

b. The Division 1 and 3 DGCW pumps and opposite unit Division 2 DGCW pump; or D

c. The Division 2 and 3 DGCW pumps. D

The Division 1 DGCW subsystem services its associated Diesel Generator (DG) and ECCS cubicle area coolers, and the LPCS pump motor cooler. The Division 2 DGCW subsystem services its associated DG and ECCS cubicle area cooler. The Division 3 DGCW subsystem services the High Pressure Core Spray (HPCS) DG and its associated ECCS cubicle area cooler. The opposite unit Division 2 DGCW subsystem services its associated DG for support of systems required by both units. D D

(continued)

BASES

BACKGROUND (continued)

The DGCW and the Residual Heat Removal Service Water (RHRSW) subsystems are subsystems to the Core Standby Cooling System (CSCS) – Equipment Cooling Water System (ECWS). The CSCS – ECWS consists of three independent piping subsystems corresponding to essential electrical power supply Divisions 1, 2, and 3. The CSCS – ECWS subsystems take a suction from the service water tunnel located in the Lake Screen House. Each DGCW pump auto-starts upon receipt of a diesel generator (DG) start signal when power is available to the pump's electrical bus or on start of ECCS cubicle area coolers. The Division 1 DGCW pump also auto-starts upon receipt of a start signal for the LPCS pump. Cooling water is then pumped from the service water tunnel by the DGCW pumps to the supported systems and components (i.e., the DGs, LPCS pump motor cooler, and the ECCS cubicle area coolers). After removing heat from these systems and components, the water from the DGCW subsystem is discharged to the CSCS pond (i.e., the Ultimate Heat Sink) through a discharge line that is common to the corresponding divisional discharge from the other unit. The discharge line terminates in the discharge structure at an elevation above the normal CSCS Pond level. A complete description of the DGCW System is presented in the UFSAR, Section 9.2.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The ability of the DGCW System to provide adequate cooling to the DGs, LPCS pump motor cooling coils and ECCS cubicle area cooling coils is an implicit assumption for the safety analyses presented in UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). The ability to provide onsite emergency AC power is dependent on the ability of the DGCW System to cool the DGs.

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

LCO

The Division 1, 2, and 3, and the opposite unit's Division 2 DGCW subsystems are required to be OPERABLE to ensure the effective operation of the DGs, the LPCS pump motor, and the ECCS equipment supported by the ECCS cubicle area coolers during a DBA or transient. The OPERABILITY of each DGSW

(continued)

BASES

LCO (continued)

subsystem is based on having an OPERABLE pump and an OPERABLE flow path capable of taking suction from the CSCS water tunnel and transferring cooling water to the associated diesel generator, LPCS pump motor cooling coils, and ECCS cubicle area cooling coils, as required.

An adequate suction source is not addressed in this LCO since the minimum net positive suction head of the DGCW pump and the maximum suction source temperature are covered by the requirements specified in LCO 3.7.3, "Ultimate Heat Sink (UHS)."

APPLICABILITY

In MODES 1, 2, and 3, the DGCW subsystems are required to support the OPERABILITY of equipment serviced by the DGCW subsystems and required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the DGCW subsystems are determined by the systems they support. Therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. Thus, the LCOs of the systems supported by the DGCW subsystems will govern DGCW System OPERABILITY requirements in MODES 4 and 5.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DGCW subsystem. This is acceptable, since the Required Actions for the Condition provide appropriate compensatory actions for each inoperable DGCW subsystem. Complying with the Required Actions for one inoperable DGCW subsystem may allow for continued operation, and subsequent inoperable DGCW subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

A.1

If one or more DGCW subsystems are inoperable, the associated DG(s) and ECCS components supported by the affected DGCW loop, including LPCS pump motor cooling coils or ECCS cubicle area cooling coils, as applicable, cannot perform their intended function and must be immediately declared inoperable. In accordance with LCO 3.0.6, this

(continued)

BASES

ACTIONS

A.1 (continued)

also requires entering into the Applicable Conditions and Required Actions for LCO 3.8.1, "AC Sources—Operating," and LCO 3.5.1, "Emergency Core Cooling Systems (ECCS)—Operating."

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

Verifying the correct alignment for manual, power operated, and automatic valves in each DGCW subsystem flow path provides assurance that the proper flow paths will exist for DGCW subsystem operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet be considered in the correct position provided it can be automatically realigned to its accident position, within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. 1D

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.2.2

This SR ensures that each DGCW subsystem pump will automatically start to provide required cooling to the associated DG, LPCS pump motor cooling coils, and ECCS cubicle area cooling coils, as applicable, when the associated DG starts and the respective bus is energized or on start of the applicable ECCS cubicle area cooler. For the Division 1 DGCW subsystem, this SR also ensures the DGCW pump automatically starts on receipt of a start signal for the unit LPCS pump. These starts may be performed using actual or simulated initiation signals. 1D

(continued)

BASES (continued)

LCO

OPERABILITY of the UHS is based on a maximum water temperature being supplied to the plant of 100°F and a minimum pond water level at or above elevation 690 ft mean sea level. In addition, to ensure the volume of water available in the CSCS pond is sufficient to maintain adequate long term cooling, sediment deposition (in the intake flume and in the pond) must be ≤ 1.5 ft and CSCS pond bottom elevation must be ≤ 686.5 ft. 10 10

APPLICABILITY

In MODES 1, 2, and 3, the UHS is required to be OPERABLE to support OPERABILITY of the equipment serviced by the UHS, and is required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the UHS is determined by the systems it supports. Therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. The LCOs of the systems supported by the UHS will govern UHS OPERABILITY requirements in MODES 4 and 5.

ACTIONS

A.1

If the CSCS pond is inoperable, due to sediment deposition > 1.5 ft (in the intake flume, CSCS pond, or both) or the pond bottom elevation > 686.5 ft, action must be taken to restore the inoperable UHS to an OPERABLE status within 90 days. The 90 day Completion Time is reasonable based on the low probability of an accident occurring during that time, historical data corroborating the low probability of continued degradation (i.e., further excessive sediment deposition or pond bottom elevation changes) of the CSCS pond during that time, and the time required to complete the Required Action.

B.1 and B.2

If the CSCS pond cannot be restored to OPERABLE status within the associated Completion Time, or the CSCS pond is determined inoperable for reasons other than Condition A (e.g., inoperable due to the temperature of the cooling water supplied to the plant from the CSCS pond > 100°F, corrected for sediment level and time of day), the unit must 10 10

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

be placed 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 3.7.3.1

Verification of the temperature of the water supplied to the plant from the CSCS pond ensures that the heat removal capabilities of the RHRSW System and DGCW System are within the assumptions of the DBA analysis. To ensure that the maximum design temperature (100°F) of water supplied to the plant is not exceeded, the temperature during normal plant operation must be ≤ 100°F, corrected for sediment level and time of day the measurement is taken (Ref. 3). This is to account for the CSCS pond design requirement that it provide adequate cooling water supply to the plant (i.e., temperature ≤ 100°F) for 30 days without makeup, while taking into account solar heat loads and plant decay heat during the worst historical weather conditions. In addition, since the lake temperature follows a diurnal cycle (it heats up during the day and cools off at night), the measured temperature must be corrected for the time of day the measurement is taken. The allowable temperatures, based on the actual sediment level and the time of day the measurement is taken, have been determined by analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.3.2

This SR ensures adequate long term (30 days) cooling can be maintained, by verifying the sediment level in the intake flume and the CSCS pond is ≤ 1.5 feet. Sediment level is determined by a series of sounding cross-sections compared to as-built soundings. The 24 month Frequency is based on historical data and engineering judgement regarding sediment deposition rate.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.7.3.3 This SR ensures adequate long term (30 days) cooling can be maintained, by verifying the CSCS pond bottom elevation is ≤ 686.5 feet. The 24 month Frequency is based on historical data and engineering judgement regarding pond bottom elevation changes.
REFERENCES	1. Regulatory Guide 1.27, Revision 2, January 1976. 2. UFSAR, Section 9.2.1. 3. UFSAR, Section 9.2.6.

DISCUSSION OF CHANGES
ITS: 3.7.3 - ULTIMATE HEAT SINK (UHS)

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 A new Surveillance Requirement (ITS SR 3.7.3.1) is added to CTS 4.7.1.3 to require verification that the temperature of the cooling water supplied to the plant from the UHS (CSCS pond) is ≤ 100°F every 24 hours. This limit on cooling water temperature being supplied from the UHS during unit operation ensures that the maximum temperature of the water to CSCS equipment assumed in the LaSalle design basis accident (DBA) analysis is not exceeded. The addition of this Surveillance Requirement represents an additional restriction on plant operation necessary to help ensure the OPERABILITY of the UHS and the heat removal capabilities of the Residual Heat Removal Service Water System and the Diesel Generator Cooling Water System are maintained within the assumptions of the DBA analyses. 1A10

When the CSCS pond is inoperable, the Action of CTS 3.7.1.3 provides a 90 day period to restore the CSCS pond to OPERABLE status. In ITS 3.7.3, the 90 day period for restoration of the CSCS pond has been maintained when the inoperability is due to sediment deposition exceeding the required limit or pond bottom depth exceeding the limit. For other inoperabilities of the CSCS pond (e.g., average water temperature not within limit), ITS 3.7.3, Required Action B.1 and B.2 will require the plant to be in MODE 3 within 12 hours and in MODE 4 within 36 hours. This change to the actions associated with an inoperable CSCS pond represents an additional restriction on operation necessary to help ensure that actions taken in the event of a loss of function associated with the Ultimate Heat Sink are maintained consistent with the actions required for a loss of function associated with the systems and components supported by the CSCS pond.

<CTS>

<LCO 3.7.1.2> [1] Insert LCO

The following DGCW subsystems shall be OPERABLE:

a. Three DGCW subsystems; and

b. The opposite unit Division 2 DGCW subsystem.

Insert BWR/4 ISTS 3.7.3 (continued) [1]

DGCW [3]

DG/[1B]/SSW System 3.7.~~3~~ 2 [1]

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<4.7.1.2.a>	SR 3.7.~~3~~ 2 1 [1] Verify each ~~DG/[1B]/SSW System~~ DGCW subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days [2] [D]
<4.7.1.2.b>	SR 3.7.~~3~~ 2 [1] Verify ~~the DG [1B] SSW System~~ each DGCW pump starts automatically ~~when DG [1B] starts and energizes the respective bus~~ on each required actual or simulated initiation signal.	~~[18]~~ 24 months [3] [2] [D]

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
2 B → C. Required Action and associated Completion Time of Condition A ~~or B~~ not met. <3.7.1.3 Act a> 2	~~C.1~~ Be in MODE 3. AND (B) 2 ~~C.2~~ Be in MODE 4.	12 hours 36 hours
2 ~~OR~~ ~~Both [SSW] subsystems inoperable [for reasons other than Condition A].~~		
3 ~~OR~~ CSCS pond ~~[UHS]~~ inoperable for reasons other than Condition A.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.7.1.1~~	~~Verify the water level of each [UHS] cooling tower basin is ≥ [7.25] ft.~~	~~24 hours~~ 2
~~SR 3.7.1.2~~	~~Verify the water level [in each SSW pump well of the intake structure] is ≥ [ ] ft.~~	~~24 hours~~ 2
<Doc M.1> SR 3.7.1.3 (3.1)	Verify ~~the average~~ Cooling water temperature ~~of~~ ~~[UHS]~~ is ≤ ~~[ ]~~ °F. (100)	24 hours 3

Supplied to the plant from the CSCS pond

*(continued)

<4.7.1.3.a> SR 3.7.3.2	Verify sediment level is ≤ 1.5 ft in the intake flume and the CSCS pond.	24 months 2
<4.7.1.3.b> SR 3.7.3.3	Verify CSCS pond bottom elevation is ≤ 686.5 ft.	24 months

Insert B 3.7.2 BKGD

The DGCW System consists of three independent cooling water headers (Divisions 1, 2, and 3), and their associated pumps, valves, and instrumentation. The pump and header for the Division 1 DGCW subsystem is common to both units (and supplies cooling to equipment on both units). The other divisions have independent pumps and suction headers.

The following combinations of DGCW pumps are sized to provide sufficient cooling capacity to support the required safety related systems during safe shutdown of the unit following a loss of coolant accident (LOCA):

a. The Division 1 and 2 DGCW pumps; D

b. The Division 1 and 3 DGCW pumps and opposite unit's Division 2 DGCW pump; or D

c. The Division 2 and 3 DGCW pumps. D

The Division 1 DGCW subsystem services its associated Diesel Generator (DG) and ECCS cubicle area coolers, and the LPCS pump motor cooler. The Division 2 DGCW subsystem services its associated DG and ECCS cubicle area cooler. The Division 3 DGCW subsystem services the High Pressure Core Spray (HPCS) DG and its associated ECCS cubicle area cooler. The opposite unit Division 2 DGCW subsystem services its associated DG for support of systems required by both units. D

The DGCW and the Residual Heat Removal Service Water (RHRSW) subsystems are subsystems to the Core Standby Cooling System (CSCS) – Equipment Cooling Water System (ECWS). The CSCS – ECWS consists of three independent piping subsystems corresponding to essential electrical power supply Divisions 1, 2, and 3. The CSCS – ECWS subsystems take a suction from the service water tunnel located in the Lake Screen House. Each DGCW pump auto-starts upon receipt of a diesel generator (DG) start signal when power is available to the pump's electrical bus or on start of ECCS cubicle area coolers. The Division 1 DGCW pump also auto-starts upon receipt of a start signal for the LPCS pump. Cooling water is then pumped from the service water tunnel by the DGCW pumps to the supported systems and components (i.e., the DGs, LPCS pump motor cooler, and the ECCS cubicle area coolers). After removing heat from these systems and components, the water from the DGCW subsystem is discharged to the CSCS pond (i.e., the Ultimate Heat Sink) through a discharge line that is common to the corresponding divisional discharge from the other unit. The discharge line terminates in the discharge structure at an elevation above the normal CSCS Pond level.

Insert B 3.7.2 LCO

The Division 1, 2, and 3, and the opposite unit's Division 2 DGCW subsystems are required to be OPERABLE to ensure the effective operation of the DGs, the LPCS pump motor, and the ECCS equipment supported by the ECCS cubicle area coolers during a DBA or transient. D

Insert BWR/4 ISTS B 3.7.3 [1]

(continued)

BASES

ACTIONS

B.1 (continued)

not restored to OPERABLE status within 60 days, DG [1B] must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

Verifying the correct alignment for manual, power operated, and automatic valves in the DG [1B] SSW System flow path provides assurance that the proper flow paths will exist for DG [1B] SSW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet be considered in the correct position provided it can be automatically realigned to its accident position, within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.3.2

This SR ensures that the DG [1B] SSW System pump will automatically start to provide required cooling to the DG [1B] when the DG [1B] starts and the respective bus is energized.

Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency, which is based at the refueling cycle. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

or on start of the applicable ECCS cubicle area cooler. For the Division 1 DGCW subsystem, this SR also ensures the DGCW pump automatically starts on receipt of a start signal for the unit LPCS pump.

These starts may be performed by using actual or simulated initiation signals.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~Is the failure of one of the two standby DGs, which would in turn affect one [SSW] subsystem. The [SSW] flow assumed in the analyses is [7900] gpm per pump to the heat exchanger (FSAR, Table [6.2-2], Ref. 7). Reference 2 discusses [SSW] System performance during these conditions.~~

The [SSW] System, together with the [UHS], satisfy Criterion 3 of the NRC Policy Statement.

LCO

~~The OPERABILITY of subsystem A (Division 1) and subsystem B (Division 2) of the [SSW] System is required to ensure the effective operation of the RHR System in removing heat from the reactor, and the effective operation of other safety related equipment during a DBA or transient. Requiring both subsystems to be OPERABLE ensures that either subsystem A or B will be available to provide adequate capability to meet cooling requirements of the equipment required for safe shutdown in the event of a single failure.~~

~~A subsystem is considered OPERABLE when:~~

~~a. The associated pump is OPERABLE;~~

~~b. The associated [UHS] is OPERABLE; and~~

~~c. The associated piping, valves, instrumentation, and controls required to perform the safety related function are OPERABLE.~~

OPERABILITY of the [UHS] is based on a maximum water temperature of [95]°F with OPERABILITY of each subsystem requiring a minimum basin water level at or above elevation [130 ft 3 inches] mean sea level (equivalent to an indicated level of ≥ [7 ft 3 inches]) and four OPERABLE cooling tower fans.

~~The isolation of the [SSW] System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the [SSW] System.~~

~~OPERABILITY of the High Pressure Core Spray (HPCS) Service Water System (SWS) is addressed by LCO 3.7.2, "HPCS SWS."~~

(continued)

In addition, to ensure the volume of water available in the CSCS pond is sufficient to maintain adequate long term cooling, sediment deposition (in the intake flume and in the pond) must be ≤ 1.5 ft and CSCS pond bottom elevation must be ≤ 686.5 ft.

BASES

ACTIONS (continued)

[C.1 and C.2]

If the [SSW] subsystem cannot be restored to OPERABLE status within the associated Completion Time, or both [SSW] Subsystems are inoperable [for reasons other than Condition A], or the [UHS] is determined inoperable for reasons other than Condition A, the unit must be placed 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.

(e.g., inoperable due to the temperature of the cooling water supplied to the plant from the CSCS pond > 100°F, corrected for sediment level at time of day)

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR ensures adequate long term (30 days) cooling can be maintained. With the [UHS] water source below the minimum level, the affected [SSW] subsystem must be declared inoperable. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.1.2

This SR verifies the water level [in each [SSW] pump well of the intake structure] to be sufficient for the proper operation of the [SSW] pumps (net positive suction head and pump vortexing are considered in determining this limit). The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.1.3

Verification of the [UHS] temperature ensures that the heat removal capability of the [SSW] System is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

(RHRSW System and DGCW System are)

INSERT SR 3.7.3.1

(continued)

Insert SRs 3.7.3.2 and 3.7.3.3

1

Insert SR 3.7.3.1

To ensure that the maximum design temperature (100°F) of water supplied to the plant is not exceeded, the temperature during normal plant operation must be ≤ 100°F, corrected for sediment level and time of day the measurement is taken (Ref. 3). This is to account for the CSCS pond design requirement that it provide adequate cooling water supply to the plant (i.e., temperature ≤ 100°F) for 30 days without makeup, while taking into account solar heat loads and plant decay heat during the worst historical weather conditions. In addition, since the lake temperature follows a diurnal cycle (it heats up during the day and cools off at night), the measured temperature must be corrected for the time of day the measurement is taken. The allowable temperatures, based on the actual sediment level and the time of day the measurement is taken, have been determined by analysis.

D

1

Insert SRs 3.7.3.2 and 3.7.3.3

SR 3.7.3.2

This SR ensures adequate long term (30 days) cooling can be maintained, by verifying the sediment level in the intake flume and the CSCS pond is ≤ 1.5 feet. Sediment level is determined by a series of sounding cross-sections compared to as-built soundings. The 24 month Frequency is based on historical data and engineering judgement regarding sediment deposition rate.

SR 3.7.3.3

This SR ensures adequate long term (30 days) cooling can be maintained, by verifying the CSCS pond bottom elevation is ≤ 686.5 feet. The 24 month Frequency is based on historical data and engineering judgement regarding pond bottom elevation changes.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two required Division 1, 2, or 3 DGs inoperable. OR Division 2 DG and the required opposite unit Division 2 DG inoperable.	F.1 Restore one required DG to OPERABLE status.	2 hours OR 72 hours if Division 3 DG is inoperable
G. Required Action and associated Completion Time of Condition A, C, D, E, or F not met. OR Required Action B.2, B.3, or B.4 and associated Completion Time not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	12 hours 36 hours
H. Three or more required AC sources inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.19	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ----------	
	Verify, on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses for Divisions 1 and 2 only; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 13 seconds, 2. energizes auto-connected emergency loads including through time delay relays, where applicable, 3. maintains steady state voltage ≥ 4010 V and ≤ 4310 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	24 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	----------NOTE---------- All DG starts may be preceded by an engine prelube period. ---------- Verify, when started simultaneously from standby condition, each required DG achieves, in ≤ 13 seconds, voltage ≥ 4010 V and frequency ≥ 58.8 Hz.	10 years D
SR 3.8.1.21	----------NOTE---------- When the opposite unit is in MODE 4 or 5, or moving irradiated fuel assemblies in secondary containment, the following opposite unit SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.9 through SR 3.8.1.11, SR 3.8.1.14 through SR 3.8.1.16. ---------- For required opposite unit DG, the SRs of the opposite unit's Specification 3.8.1, except SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.17, SR 3.8.1.18, SR 3.8.1.19, and SR 3.8.1.20, are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil and Starting Air

LCO 3.8.3 The stored diesel fuel oil and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with stored fuel oil level: 1. In the fuel oil storage tank for the Division 1 and Division 2 DGs, and the opposite unit Division 2 DG, < 31,000 gal and ≥ 26,550 gal; and 2. In the combined day tank and fuel oil storage tank for the Division 3 DG, < 29,750 gal and ≥ 25,550 gal.	A.1 Restore stored fuel oil level to within limit.	48 hours

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources — Shutdown

LCO 3.8.5 The following DC electrical power subsystem(s) shall be OPERABLE:

a. One Division 1 125 VDC or Division 2 125 VDC electrical power subsystem capable of supplying one division of the onsite Class 1E DC Electrical Power Distribution System required by LCO 3.8.8, "Distribution Systems — Shutdown";

b. The Division 3 125 VDC electrical power subsystem, when the Division 3 onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.8; and

c. The opposite unit Division 2 125 VDC electrical power distribution subsystem, when the opposite unit Division 2 onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.8.

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. ---------NOTE--------- Not applicable when the opposite unit is in MODE 1, 2, or 3. ---------- One or more required Division 1, 2, and 3 DC electrical power subsystems inoperable.	A.1 Verify associated DC electrical power distribution subsystem is energized by OPERABLE opposite unit DC electrical power subsystem.	1 hour
	AND	
	A.2 Restore required Division 1, 2, and 3 DC electrical power subsystem to OPERABLE status.	72 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Declare affected required feature(s) inoperable.	Immediately
OR	OR	
Required opposite unit Division 2 DC electrical power subsystem inoperable.	B.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
OR	B.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
---------NOTE--------- Only applicable when the opposite unit is in MODE 1, 2, or 3. ----------	AND	
One or more required Division 1, 2, and 3 DC electrical power subsystems inoperable.	B.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	B.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	----------NOTE---------- The following SRs are not required to be performed: SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. ---------- For DC electrical power subsystems required to be OPERABLE the following SRs are applicable: SR 3.8.4.1, SR 3.8.4.2, SR 3.8.4.3, SR 3.8.4.4, SR 3.8.4.5, SR 3.8.4.6, SR 3.8.4.7, SR 3.8.4.8, and SR 3.8.4.9	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the Division 1, 2, and 3 and opposite unit Division 2 batteries shall be within limits.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category A or B limits.	A.1	Verify pilot cells electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	AND		
	A.2	Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours AND Once per 7 days thereafter
	AND		
	A.3	Restore battery cell parameters to Table 3.8.6-1 Category A and B limits.	31 days

(continued)

BASES

LCO (continued)

In addition, day tank storage and fuel oil transfer system requirements must be met for each required DG.

The AC sources in one division must be separate and independent (to the extent possible) of the AC sources in the other division(s). For the DGs, the separation and independence are complete. For the offsite AC sources, the separation and independence are to the extent practical. A qualified circuit may be connected to all divisions of either unit, with manual transfer capability to the other circuit OPERABLE, and not violate separation criteria. A qualified circuit that is not connected to the 4.16 kV emergency buses is required to have OPERABLE manual transfer capability (from the control room) to the associated 4.16 kV emergency buses to support OPERABILITY of that qualified circuit.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Note 1 has been added taking exception to the Applicability requirements for Division 3 sources, provided the High Pressure Core Spray (HPCS) System is declared inoperable. This exception is intended to allow declaring of the Division 3 inoperable either in lieu of declaring the Division 3 source inoperable, or at any time subsequent to entering ACTIONS for an inoperable Division 3 source. This exception is acceptable since, with the Division 3 inoperable and the associated ACTIONS entered, the Division 3 AC sources provide no additional assurance of meeting the above criteria. In addition, when this Note allowance is being used, both AC sources could be inoperable such that the Division 3 AC distribution subsystem is de-energized. In this case (the Division 3 AC electrical power distribution subsystem inoperable), LCO 3.0.6 would not preclude entry into the Distribution System ACTIONS since, with the Division 3 AC sources not required OPERABLE as

(continued)

BASES

APPLICABILITY (continued)

allowed by this Note, the Division 3 AC sources cannot be considered as a support system to the Division 3 AC distribution subsystem. Thus, as required by LCO 3.0.2, the Distribution System-Operating ACTIONS for the inoperable Division 3 AC electrical power distribution subsystem must be entered.

Note 2 has been added taking exception to the Applicability requirements for the required opposite unit's Division 2 DG in LCO 3.8.1.c, provided the associated required equipment is inoperable (i.e., one SGT subsystem, one primary containment hydrogen recombiner subsystem, one control room area filtration subsystem, and one control room area ventilation air conditioning subsystem). This exception is intended to allow declaring the opposite unit's Division 2 supported equipment inoperable either in lieu of declaring the opposite unit's Division 2 DG inoperable, or at any time subsequent to entering ACTIONS for an inoperable opposite unit Division 2 DG. This exception is acceptable since, with the opposite unit powered Division 2 equipment inoperable and the associated ACTIONS entered, the opposite unit Division 2 DG provides no additional assurance of meeting the above criteria.

AC power requirements for MODES 4 and 5 and other conditions in which AC sources are required are covered in LCO 3.8.2, "AC Sources — Shutdown."

ACTIONS

A.1

To ensure a highly reliable power source remains, it is necessary to verify the availability of the remaining required offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in the Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition D, for two required offsite circuits inoperable, is entered.

A.2

Required Action A.2, which only applies if the division cannot be powered from an offsite source, is intended to

(continued)

BASES

ACTIONS

A.2 (continued)

provide assurance that an event with a coincident single failure of the associated DG does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions (i.e., single division systems are not included, although, for this Required Action, Division 3 (HPCS System) is considered redundant to Division 1 and 2 ECCS). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has no offsite power available.

The Completion Time for Required Action A.2 is intended to allow time for the operator to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The division has no offsite power available to supply its loads; and

b. A redundant required feature on another division is inoperable.

If, at any time during the existence of this Condition (one required offsite circuit inoperable), a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power available to one division of the onsite Class 1E Power Distribution System coincident with one or more inoperable redundant required support or supported features, or both, that are associated with the other division that has offsite power, results in starting the Completion Time for the Required Action.
Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before the unit is subjected to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection may have been lost for the required

(continued)

BASES

ACTIONS

A.2 (continued)

feature's function; however, function is not lost. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. C

With one required offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the plant safety systems. In this condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E distribution system.

The Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period. C

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, the common DG is inoperable for pre-planned maintenance and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total of 10 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a unit DG could again become inoperable, the circuit restored OPERABLE, and an additional 72 hours (for a total of 13 days) allowed prior to complete restoration of the LCO. The 10 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet LCO 3.8.1.a or b. This limit is considered C C C

(continued)

BASES

ACTIONS

A.3 (continued)

reasonable for situations in which Conditions are entered concurrently for combinations of Conditions A, B, and C. The "AND" connector between the 72 hour and 10 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive must be met.

Similar to Required Action A.2, the Completion Time of Required Action A.3 allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This exception results in establishing the "time zero" at the time LCO 3.8.1.a or b was initially not met, instead of at the time that Condition A was entered.

B.1

Condition B provides appropriate compensatory measures to allow performance of pre-planned maintenance or testing on the common DG. Pre-planned maintenance or testing includes preventative maintenance, modifications, and performance of Surveillance Requirements. The Note effectively only allows Condition B to be used for the common DG when the opposite unit is not in MODE 1, 2, or 3. When the common DG becomes inoperable while both units are in MODE 1, 2, or 3, Condition C must be entered for both units and the associated Required Actions performed.

Required Action B.1, is intended to provide assurance that a loss of offsite power, during the period that the common DG or its supported equipment is inoperable for the purposes of completing pre-planned maintenance, modifications, or Surveillance Requirements, does not result in a complete loss of safety function of critical systems. This is accomplished by making an additional source available to support the unit and opposite unit Division 2 emergency buses. This additional source is the unit or opposite unit Division 2 DG. To ensure this alternate highly reliable power source is available during operation in Condition B, it is necessary to temporarily modify the control circuit for the unit crosstie circuit breakers between 4.16 kV emergency buses 142Y and 242Y to allow the breakers to be closed with a DG powering one of the Division 2 emergency buses (142Y or 242Y) so that the unit or opposite unit Division 2 DG can supply the unit and opposite unit

(continued)

BASES

ACTIONS

B.1 (continued)

Division 2 emergency buses. Therefore, the unit or opposite unit Division 2 DG must be OPERABLE with the capability to be manually aligned to the unit and opposite unit Division 2 emergency buses. The Completion Time ensures the alternate source to the Division 2 emergency buses is available whenever the plant is operating in Condition B. If Required Action B.1 and the associated Completion Time are not met, Condition C must the entered and the Required Actions taken.

B.2

To ensure a highly reliable power source remains, it is necessary to verify the availability of the remaining required offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure to meet SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions must then be entered.

B.3

Required Action B.3 is intended to provide assurance that a loss of offsite power, during the period that the common DG is inoperable for the purposes of completing pre-planned maintenance, modifications, or Surveillance Requirements on the common DG or its support systems, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions (i.e., single division systems are not included, although for this Required Action, Division 3 (HPCS) is considered redundant to Division 1 and Division 2 ECCS). Redundant required feature failures consist of inoperable features associated with a division redundant to the division that has an inoperable DG.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

(continued)

BASES

ACTIONS

B.3 (continued)

a. An inoperable common DG exists; and

b. A redundant required feature on another division is inoperable.

If, at any time during the existence of this Condition (the common DG inoperable due to pre-planned maintenance, modification, or testing), a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering the common DG inoperable coincident with one or more redundant required support or supported features, or both, that are associated with the redundant OPERABLE DG(s), results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown. The remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and low probability of a DBA occurring during this period.

B.4

One common DG provides onsite standby power to the Division 1 emergency buses on both units. This Required Action provides a 7 day time period to perform pre-planned maintenance or testing on the common DG while precluding the shutdown of both units. Pre-planned maintenance or testing includes preventative maintenance, modifications, and performance of Surveillance Requirements. The Note to Condition B effectively only allows the 7 day Completion Time to be used for the common DG when the opposite unit is not in MODE 1, 2, or 3. When the common DG becomes

(continued)

BASES

ACTIONS

B.4 (continued)

inoperable while both units are in MODE 1, 2, or 3, Condition C must be entered for both units and the associated Required Actions performed. The 4.16 kV emergency bus design is sufficient to allow operation to continue in Condition B for a period that should not exceed 7 days. In this condition, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet LCO 3.8.1.a or b. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This situation could lead to a total of 10 days, since initial failure of the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 13 days) allowed prior to complete restoration of the LCO. The 10 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet LCO 3.8.1.a or b. This limit is considered reasonable for situations in which Conditions are entered concurrently for combinations of Conditions A, B, and C. The "AND" connector between the 7 day and 10 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive must be met.

Similar to Required Action B.3, the Completion Time of Required Action B.4 allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This exception results in establishing the "time zero" at the time LCO 3.8.1.a or b was initially not met, instead of the time that Condition B was entered.

(continued)

BASES

ACTIONS (continued)

C.1

To ensure a highly reliable power source remains, it is necessary to verify the availability of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions must then be entered.

C.2

Required Action C.2 is intended to provide assurance that a loss of offsite power, during the period that the DG(s) is inoperable as described in Condition C, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions (i.e., single division systems are not included, although, for this Required Action, Division 3 (HPCS System) is considered redundant to Division 1 and 2 ECCS). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has an inoperable DG.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A redundant required feature on another division is inoperable.

If, at any time during the existence of this Condition (DG(s) inoperable as described in Condition C), a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering required DG(s) inoperable coincident with one or more redundant required support or supported features, or

(continued)

BASES

ACTIONS

C.2 (continued)

both, that are associated with the redundant OPERABLE DG(s), results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

C.3.1 and C.3.2

Required Action C.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG(s) does not exist on the OPERABLE DG(s), SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DGs, the other DGs are declared inoperable upon discovery, and Condition F or H of LCO 3.8.1 is entered, as applicable. Once the failure is repaired, and the common cause failure no longer exists, Required Action C.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG(s), performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of those DG(s).

In the event the inoperable DG(s) is restored to OPERABLE status prior to completing either C.3.1 or C.3.2, the station corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition C.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

continues to envelope the duty cycle of the service test.) Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test. C D

A modified performance discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test when the modified performance discharge test is performed in lieu of a service test. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

The acceptance criteria for this Surveillance is consistent with IEEE-450 (Ref. 8) and IEEE-485 (Ref. 11). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating, since IEEE-485 (Ref. 11) recommends using an ageing factor of 125% in the battery sizing calculation. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturers rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

manufacturers rating. Degradation is indicated, consistent with IEEE-450 (Ref. 8), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturers rating. The 12 month and 60 month Frequencies are consistent with the recommendations in IEEE-450 (Ref. 8). The 24 month Frequency is derived from the recommendations of IEEE-450 (Ref. 8).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required 125 VDC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR. C

SR 3.8.4.9

With the exception of this Surveillance, all other Surveillances of this Specification (SR 3.8.4.1 through 3.8.4.8) are applied to the given unit DC sources. This Surveillance is provided to direct that appropriate Surveillances for the required opposite unit DC source are governed by the applicable opposite unit Technical

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.9 (continued)

Specifications. Performance of the applicable opposite unit Surveillances will satisfy the opposite unit requirements as well as satisfy the given unit Surveillance Requirement.

The Frequency required by the applicable opposite unit SR also governs performance of that SR for the given unit.

As noted, if the opposite unit is in MODE 4 or 5, or moving irradiated fuel assemblies in secondary containment, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8 are not required to be performed. This ensures that a given unit SR will not require an opposite unit SR to be performed, when the opposite unit Technical Specifications exempts performance of an opposite unit SR (however, as stated in the opposite unit SR 3.8.5.1 Note 1, while performance of an SR is exempted, the SR must still be met).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE Standard 308, 1971.
4. UFSAR, Section 8.3.2.
5. UFSAR, Chapter 6.
6. UFSAR, Chapter 15.
7. Regulatory Guide 1.93, December 1974.
8. IEEE Standard 450, 1995. |P
9. Regulatory Guide 1.32, August 1972.
10. IEEE Standard 485, 1978.

BASES

LCO (continued)

inoperable due to a failure also affecting the OPERABILITY of a bus listed in Table B 3.8.7-1 for Unit 1 and Table B 3.8.7-2 for Unit 2 (e.g., loss of 4.16 kV emergency bus, which results in de-energization of all buses powered from the 4.16 kV emergency bus), then although the individual loads are still considered inoperable, the Conditions and Required Actions of the LCO for the individual loads are not required to be entered, since LCO 3.0.6 allows this exception (i.e., the loads are inoperable due to the inoperability of a support system governed by a Technical Specification; the 4.16 kV emergency bus).

In addition, at least one tie breaker between the redundant Division 2, safety related DC emergency power distribution subsystems must be open. This prevents an electrical malfunction in one power distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s). If at least one tie breaker is not open, then both Division 2 DC electrical power distribution subsystems are considered inoperable. The restriction of maintaining electrical separation applies to the onsite, safety related, redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 4.16 kV emergency buses from being supplied from the same offsite source.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained, in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 4 and 5 and other conditions in which AC and DC electrical power distribution subsystems are required. are covered in the Bases for LCO 3.8.8, "Distribution Systems—Shutdown."

(continued)

BASES

ACTIONS

A.1

With one or more Division 1 and 2 required AC buses, load centers, motor control centers, or distribution panels inoperable and a loss of function has not yet occurred, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels must be restored to OPERABLE status within 8 hours.

The Condition A worst scenario is two divisions without AC power (i.e., no offsite power to the divisions and the associated DGs inoperable). In this situation, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operators' attention be focused on minimizing the potential for loss of power to the remaining division by stabilizing the unit and restoring power to the affected division. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the unit operators' attention is diverted from the evaluations and actions necessary to restore power to the affected division to the actions associated with taking the unit to shutdown within this time limit.

b. The low potential for an event in conjunction with a single failure of a redundant component in the division with AC power. (The redundant component is verified OPERABLE in accordance with Specification 5.5.12, "Safety Function Determination Program (SFDP).")

(continued)

BASES

ACTIONS

A.1 (continued)

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet LCO 3.8.7.a. If Condition A is entered while, for instance, a DC electrical power distribution subsystem is inoperable and subsequently returned OPERABLE, LCO 3.8.7.a may already have been not met for up to 2 hours. This situation could lead to a total duration of 10 hours, since initial failure of LCO 3.8.7.a, to restore the AC electrical power distribution system. At this time, a DC electrical power distribution subsystem could again become inoperable, and the AC electrical power distribution could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This results in establishing the "time zero" at the time LCO 3.8.7.a was initially not met, instead of at the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet LCO 3.8.7.a indefinitely.

B.1

With one or more Division 1 and 2 DC electrical distribution subsystems inoperable and a loss of function has not yet occurred, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required DC electrical power distribution subsystem(s) must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

(continued)

BASES

ACTIONS

B.1 (continued)

Condition B worst scenario is two divisions without adequate DC power, potentially with both the battery significantly degraded and the associated charger nonfunctioning. In this situation, the plant is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the plant, minimizing the potential for loss of power to the remaining divisions, and restoring power to the affected division(s).

This 2 hour limit is more conservative than Completion Times allowed for the majority of components that could be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, that would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety when requiring a change in plant conditions (i.e., requiring a shutdown) while not allowing stable operations to continue;

b. The potential for decreased safety when requiring entry into numerous applicable Conditions and Required Actions for components without DC power while not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected division; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC electrical power distribution subsystems is consistent with Regulatory Guide 1.93 (Ref. 3).

(continued)

A.1

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.1.19

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 13 seconds, energizes the auto-connected emergency loads through the load sequencer (time delay relays) and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ±~~416~~ (150) volts and 60 ±1.2 Hz during this test. A.19 M.11 C B D

b) For Division 3:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with its loads within 13 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ±~~416~~ (150) volts and 60 ±1.2 Hz during this test. M.1 M.11 B D

SR 3.8.1.13

7. Verifying that all diesel generator 0, 1A, and 1B automatic trips except the following are automatically bypassed on an ECCS actuation signal: (actual or simulated) L.12

a) For Divisions 1 and 2 - engine overspeed, generator differential current, and emergency manual stop. A.13 A.11

b) For Division 3 - engine overspeed, generator differential current, and emergency manual stop. A.12

SR 3.8.1.14

8. Verifying the diesel generator operates (Add proposed SR 3.8.1.14 Note 4) (Add power factor requirement) for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 2860 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to 2400 kW to 2600 kW.*** ~~The generator voltage and frequency shall be 4160 +420, -150 volts and 60 +3.0, -1.2 Hz within 13 seconds after the start signal; the steady state~~ M.10 L.13 C

A.12

~~*All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, as recommended by the manufacturer.~~

SR 3.8.1.14 Note 1 SR 3.8.1.15 Note 1 ***Transients, outside of this load band, do not invalidate the surveillance tests.

A.1

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.1.19

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 13 seconds, energizes the auto-connected emergency loads through the load sequencer (time delay relays) and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ±~~415~~ (150) volts and 60 ±1.2 Hz during this test. A.19 M.11 C D

b) For Division 3:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with its loads within 13 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ±~~415~~ (150) volts and 60 ±1.2 Hz during this test. M.1 M.11 D

SR 3.8.1.13

7. Verifying that all diesel generator 0, 2A, and 2B automatic trips except the following are automatically bypassed on an actual or simulated ECCS actuation signal: L.12

a) For Divisions 1 and 2 - engine overspeed, generator differential current, ~~and emergency manual stop~~ A.13 A.11

b) For Division 3 - engine overspeed, generator differential current, ~~and emergency manual stop~~ A.12

SR 3.8.1.14

8. Add proposed SR 3.8.1.14 Note 4 — Add power factor requirement — Verifying the diesel generator operates* for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 2860 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to 2400 kW to 2600 kW.*** ~~The generator voltage and frequency shall be 4160 +420, -150 volts and 60 +3.0, -1.2 Hz within 13 seconds after the start signal; the steady-state~~ M.10 L.13 C

~~*All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, as recommended by the manufacturer.~~ A.12

SR 3.8.1.14 Note 1
SR 3.8.1.15 Note 1
***Transients, outside of this load band, do not invalidate the surveillance tests.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.9 (cont'd) adversely affect one DG. Proposed SR 3.8.1.3 Note 4 requires that this SR be immediately preceded by a successful performance of SR 3.8.1.2 (the DG start Surveillance). This will ensure the DG load carrying capability is tested subsequent to a successful DG start test. While these Notes clearly represent current LaSalle 1 and 2 practice, they are more restrictive than the CTS since the SR could currently be performed without these restrictions.

M.10 Limitations on the operating power factor are added to CTS 4.8.1.1.2.d.8, the 24-hour run Surveillance (proposed SR 3.8.1.14, including Note 3). These limitations ensure the DG is conservatively tested at as close to accident conditions as reasonable provided the power factor can be attained. The actual power factor values have been added to the Bases. A Note has been also added to CTS 4.8.1.1.2.d.8 (proposed SR 3.8.1.14 Note 1) to ensure a momentary transient that results in the power factor not being met does not invalidate the 24 hour run. The change to include any power factor requirement is more restrictive on plant operation. C

M.11 CTS 4.8.1.1.2.d.7 requires the steady state voltage to be 4160 $\pm$ 416 V. Proposed SR 3.8.1.19 requires the steady state voltage to be 4160 $\pm$ 150 V. CTS 4.8.1.1.2.e, the 10 year DG simultaneous start test, does not provide a minimum voltage the DGs must attain within the 13 second DG start time assumed in the accident analysis. Proposed SR 3.8.1.20 requires the minimum voltage to be 4010 V. The new voltage limits ensure that components powered by the associated bus will have sufficient voltage to perform their required function. These acceptance criteria are consistent with all other DG start acceptance criteria. These are added restrictions on plant operation. D D

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS 3.8.1.1.a, 3.8.1.1.b, 3.8.1.1.b.3, and 3.8.2.1.d details relating to system design and OPERABILITY (i.e., that the offsite circuits are "physically independent," the DGs are "separate and independent," the nomenclature of the DGs, that each DG has "a separate fuel transfer pump," and some components of the opposite unit's offsite circuit) are proposed to be relocated to the Bases. The details for system OPERABILITY are not necessary in the LCO. The definition of OPERABILITY suffices. The design details are not necessary to be included in the Technical Specifications to ensure the OPERABILITY of the AC Sources

capable of supplying one division of the onsite Class 1E DC Electrical Power Distribution System required by LCO 3.8.8, "Distribution Systems - Shutdown," M.1

ITS 3.8.5

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - SHUTDOWN

A.1

LIMITING CONDITION FOR OPERATION

<General Description> A.2

D

LCO 3.8.5 3.8.2.4 As a minimum, Division 1 or Division 2, and Division 3 when the HPCS system is required to be OPERABLE, and Unit 2 Division 2 when the standby gas treatment system and/or the control room and auxiliary electric equipment room emergency filtration system are required to be OPERABLE, of the D.C. distribution system shall be OPERABLE and energized with:

C

a. Division 1, consisting of;
   1. 125 volt battery 1A.
   2. 125 volt full capacity charger.
   3. 125 volt distribution panel 111Y.

b. Division 2, consisting of;
   1. 125 volt battery 1B.
   2. 125 volt full capacity charger.
   3. 125 volt distribution panel 112Y.

c. Division 3, consisting of;
   1. 125 volt battery 1C.
   2. 125 volt full capacity charger.
   3. 125 volt distribution panel 113.

d. Unit 2 Division 2, consisting of;
   1. 125 volt battery 2B.
   2. 125 volt full capacity charger.
   3. 125 volt distribution panel 212Y.

LA.1

See ITS 3.8.8

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

C

add proposed Required Action B.1 L.3

ACTION B

a. With both Division 1 distribution panel 111Y and Division 2 distribution panel 112Y of the above required D.C. distribution system inoperable or not energized, suspend CORE ALTERATIONS, handling of irradiated fuel cask in the secondary containment and operations with a potential for draining the reactor vessel.

See ITS 3.8.8

ACTION B

b. With Division 3 distribution panel 113 of the above required D.C. distribution system inoperable or not energized, declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3. A.3

add proposed Required Actions B.2.1 and B.2.2 M.3

add proposed Required Action B.2.4 M.2

*When handling irradiated fuel in the secondary containment.

Applicability

Page 1 of 4

capable of supplying one division of the onsite Class 1E DC Electrical Power Distribution System required by LCO 3.8.8, "Distribution Systems - Shutdown,"

M.1 D

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - SHUTDOWN

A.1

LIMITING CONDITION FOR OPERATION <General Description> A.2

LCO 3.8.5 **3.8.2.4 As a minimum, Division 1 or Division 2, and Division 3 when the HPCS system is required to be OPERABLE, and Unit 1 Division 2 when the standby gas treatment system and/or the control room and auxiliary electric equipment room emergency filtration system are required to be OPERABLE, of the D.C. distribution system shall be OPERABLE and energized with:** C

a. Division 1, consisting of:

1. 125-volt battery 2A.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 211Y.

b. Division 2, consisting of:

1. 125-volt battery 2B.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 212Y.

LA.1

c. Division 3, consisting of:

1. 125-volt battery 2C.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 213.

See ITS 3.8.8

d. Unit 1 Division 2, consisting of:

1. 125-volt battery 1B.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 112Y.

LA.1

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

add proposed Required Action B.1 L.3 C

ACTION B **a. With both Division 1 distribution panel 211Y and Division 2 distribution panel 212Y of the above required D.C. distribution system inoperable or not energized, suspend CORE ALTERATIONS, handling of irradiated fuel cask in the secondary containment and operations with a potential for draining the reactor vessel.** See ITS 3.8.8

ACTION B **b. With Division 3 distribution panel 213 of the above required D.C. distribution system inoperable or not energized, declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3.** A.3

add proposed Required Actions B.2.1 and B.2.2 M.3

add proposed Required Action B.2.4 M.2

*When handling irradiated fuel in the secondary containment.

APPLICABILITY

ADMINISTRATIVE

A.4 (cont'd) control room auxiliary electric equipment room emergency filtration subsystems are found to be inoperable, therefore, the addition of these changes are considered administrative. E

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The existing requirement of CTS 3.8.2.4 for "Division 1 or Division 2" DC electrical power sources to be OPERABLE during shutdown conditions is not specific as to what the single source must be powering. The requirement in ITS LCO 3.8.5 specifies that the source must be capable of supplying one division of the onsite Class 1E DC Electrical Power Distribution System required by LCO 3.8.8, "Distribution Systems—Shutdown." This added restriction conservatively assures that at least Division 1 or Division 2 DC electrical power distribution subsystem has an OPERABLE DC source (battery and associated charger) supplying it with power, when required. C D

M.2 In the event the necessary Division 1 or 2 DC source is not OPERABLE, plant conditions are conservatively restricted in CTS 3.8.2.4 Action a (ITS 3.8.5 Required Actions B.2.1, B.2.2, and B.2.3) by suspending CORE ALTERATIONS, irradiated fuel handling, and OPDRVs. In the event the necessary Division 3 DC source is not OPERABLE, plant conditions are conservatively restricted by suspending OPDRVs as required by CTS 3.8.2.4 Action b and CTS 3.5.2 Action a. However, continued operation without the necessary DC sources should not be considered acceptable. Therefore, ITS 3.8.5 Required Action B.2.4 is added to commence and continue attempts to restore the necessary DC sources. (Note that if actions are taken in accordance with ITS 3.8.5 Required Action B.1, sufficiently conservative measures are assured by the ACTIONS for the individual components declared inoperable without requiring the efforts to restore the inoperable source.) ITS 3.8.5 Required Action B.2.4 results in an action which does not allow continued operation in the existing plant condition. This has the effect of not allowing MODE changes per LCO 3.0.4. Therefore, this existing implicit requirement is explicitly addressed in the ITS 3.8.5 ACTIONS. C

M.3 In lieu of declaring the HPCS System inoperable and taking the ACTIONS of the appropriate LCO as required by CTS 3.8.2.4 Action b, new Required Actions have been provided for when the Division 3 DC source is inoperable, consistent with the current actions for inoperable Division 1 and 2 DC Sources (CTS 3.8.2.4 Action a). ITS 3.8.5 Required Actions B.2.1, B.2.2, and B.2.3 require suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies in the secondary containment, and OPDRVS. These Required

TECHNICAL CHANGES - MORE RESTRICTIVE

M.3 (cont'd) Actions are more restrictive than currently required, since CTS 3.5.2 Action a only requires OPDRVs to be suspended (and it allows 4 hours to start this action), and ensure proper actions are taken to compensate for an inoperable HPCS System.

M.4 A Note has been added to CTS 3.8.2.4 Action c (ITS 3.8.5 Condition A) to not allow the actions to be taken when the opposite unit is in MODE 1,2, or 3. With one DC electrical power source division (battery and/or battery charger inoperable) inoperable, CTS 3.8.2.4 Action c allows operation to continue for 72 hours as long as the associated 125V DC electrical power distribution subsystem is energized by the OPERABLE opposite unit DC electrical power subsystem. This allowance can not be used with the opposite unit in MODES 1, 2, and 3 since the associated subsystems are required the support the Operability of opposite unit safety equipment. The Division 2 DC electrical power source subsystem for each unit supports redundant safety equipment for both units and the batteries have insufficient capacity to support the required loads of both units if either unit is in MODES 1, 2, or 3. Therefore, this allowance is only permitted to be used when both units are in shutdown conditions (MODE 4, 5, or defueled) when divisional separation is not required.

M.5 In lieu of declaring the standby gas treatment subsystem and control room and auxiliary electric equipment room emergency filtration subsystem inoperable and taking the Actions of the appropriate LCO as required by CTS 3.8.2.4 Action d, three new Required Actions have been provided for when the opposite unit's Division 2 DC source is inoperable. ITS 3.8.5 Required Action B.2.1, B.2.2 and B.2.3 require immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies in the secondary containment, and OPDRVs. When one standby gas treatment subsystem or one control room and auxiliary electric equipment room emergency filtration subsystem is inoperable, CTS 3.6.5.3 and 3.7.2, respectively, allow 7 days to restore the associated subsystems to OPERABLE status prior to suspending CORE ALTERATIONS, movement of irradiated fuel assemblies in the secondary containment, and OPDRVs. Therefore, the addition of these Required Actions are considered more restrictive. C

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The requirements for OPERABLE DC electrical power distribution subsystems are contained in ITS 3.8.8, "Distribution Systems—Shutdown." Thus, ITS LCO 3.8.5 has been written to require the Division 1 or 2, Division 3 (when Division 3 electrical power distribution subsystem is required by LCO 3.8.8) and the opposite unit Division 2 (when opposite unit Division 2 electrical power distribution subsystem is required by LCO 3.8.8) DC electrical power distribution subsystems to be OPERABLE, and the details relating to system OPERABILITY in CTS 3.8.2.4 (what constitutes a required DC electrical power source) are proposed to be relocated to the Bases. The actual battery identification numbers are proposed to be relocated to the UFSAR. The Bases will include an adequate description of the batteries to properly identify them. The details for system OPERABILITY are not necessary in the LCO. The definition of OPERABILITY suffices. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. C

"Specific"

L.1 Three of the DC sources Surveillances required to be performed by CTS 4.8.2.4 (CTS 4.8.2.3.2.d, 4.8.2.3.2.e, and 4.8.2.3.2.f) involve tests that would cause the only required OPERABLE unit 125V battery to be rendered inoperable. This condition presents a significant risk if an event were to occur during the test. The NRC has previously provided Surveillance exceptions in the LaSalle 1 and 2 CTS to avoid a similar condition for the AC sources, but the exceptions have not been applied to DC sources. In an effort to consistently address this concern, proposed SR 3.8.5.1 has a Note that excludes performance requirements of Surveillances that would require the required OPERABLE unit 125V battery(s) to be rendered inoperable. This allowance does not take exception to the requirement for the battery to be capable of performing the particular function - just to the requirement to demonstrate that capability while that source of power is being relied on to support meeting the LCO.

L.2 With one DC electrical power division (battery and/or battery charger inoperable) inoperable, CTS 3.8.2.4 Action c allows operation to continue for 72 hours as long as the associated 125V DC electrical power distribution subsystem is energized by the OPERABLE opposite unit DC electrical power subsystem. Since the CTS allowance does not specify an explicit time period for alignment,

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 (cont'd) the time is considered as immediate. Therefore the DC electrical power distribution division would have to be declared inoperable immediately upon discovery. This time has been extended from immediately to 1 hour (ITS 3.8.4 Required Action A.1 Completion Time). The hour time period provides sufficient time to safely perform the alignment and restore power to the required equipment, while minimizing the risk associated with an event occurring during this time period which would require the affected equipment to be Operable. The change is acceptable since the time allowed is short and allows operations to concentrate on restoring power to the required equipment instead of suspending activities which would be resumed once power is restored.

L.3 An alternative is proposed in the LaSalle 1 and 2 ITS to suspending operations if a DC Source is inoperable, and movement of irradiated fuel assemblies, CORE ALTERATIONS, or OPDRVs are being conducted. The alternative, ITS 3.8.5 Required Action B.1, is to declare the affected feature(s) inoperable, and continue to conduct operations (e.g., OPDRVs), if the affected feature(s) ACTIONS allow. Conservative actions can be assured if the affected feature(s) without the necessary DC power is declared inoperable and the associated ACTIONS of the individual feature(s) taken. These conservative actions are currently approved (or will be approved by the ITS amendment) by the NRC. Therefore, this change is considered acceptable. C

RELOCATED SPECIFICATIONS

None

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.3 A new requirement has been added to CTS Table 4.8.2.3.2-1 footnotes (1) and (2) for when a Category A or B limit is not met. ITS 3.8.6 Required Action A.1 requires a check within 1 hour that the pilot cell electrolyte level and float voltage are within the Category C limits (CTS Table 4.8.2.3.2-1 Category B allowable values). This action ensures that if the pilot cell is exceeding Category C limits, the battery will be declared inoperable immediately. As such, this change is an additional restriction on plant operation.

M.4 New Surveillance Requirements have been added to CTS 4.7.3.d. Currently, CTS 4.7.3.d.1 only requires pilot cell electrolyte level and specific gravity to be checked every 7 days. ITS SR 3.8.6.1 will require all Category A limits to be verified, which includes not only electrolyte level and specific gravity, but also the individual pilot cell voltage. Also, CTS 4.7.3.d.2 does not require individual cell voltage to be checked every 92 days. ITS SR 3.8.6.2 will require all Category B limits to be verified, which include individual cell voltage. In addition, ITS SR 3.8.6.3 requires the average electrolyte temperature of representative cells to be verified $\geq$ 65°F for the 250V battery. This requirement is not currently required by CTS 4.7.3.d. These new SRs help ensure the Division 1 250 V DC battery can perform its safety function. These SRs are new restrictions on plant operation. | D

M.5 CTS 4.7.3.d.1.b) and 4.7.3.d.2.c) requires that the electrolyte level for each connected cell of the Division 1 250V battery be above the plates. ITS Table 3.8.6-1 maintains this current limit as a Category C limit, but also adds an additional restriction that the electrolyte level cannot be overflowing. If this Category C limit is exceeded, the battery will be declared inoperable immediately, consistent with the CTS. In addition, ITS Table 3.8.6-1 will apply new limits, Category A and Category B. These new limits are applicable to each connected cell, including the pilot cell. These new limits will require the electrolyte level to be greater than the minimum level indication mark and less than or equal to 1/4 inch above the maximum level indication mark. These limits are modified by ITS Table 3.8.6-1 footnote (a), which allows the limits to be exceeded during and following an equalizing charge, provided it is not overflowing. If these new limits are exceeded, ITS 3.8.6 ACTION A will require the limits to be restored within 31 days, as well as ensuring the Category C limits continue to be met during this 31 day period. If not restored, ITS 3.8.6 ACTION B requires the associated DC electrical power subsystem to be immediately declared inoperable and the appropriate ACTIONS of ITS 3.8.4 taken (i.e., RCIC and the RCIC PCIV will be declared inoperable and the ACTIONS of the individual System Specifications taken). These new restrictions ensure that the battery electrolyte level is maintained within normal parameters so that the battery can perform its intended function.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.6 (cont'd) limits are based on the recommendations of the vendor and of IEEE-450. With an individual cell voltage as low as 2.07, the battery will still be able to perform its intended function. ITS SR 3.8.4.1 will continue to require the battery terminal voltage to be verified against a limit while on float charge. Therefore, this change is considered acceptable.

In addition, CTS 4.7.3.d.2.a) requires that the voltage decrease from the value observed during the original test not exceed 12 volts and CTS 4.7.3.d.2.b) requires that the specific gravity decrease from the value observed during the previous test not exceed 0.05. These requirements are not maintained in the ITS. Degradation does not necessarily mean that the battery is inoperable; it is just indicating that the battery is aging and that its capacity is reduced. Two new Surveillances have been added that require a battery service test and a battery modified performance discharge or performance discharge test to be performed (See Discussion of Change M.2 for ITS 3.8.4). These new SRs are adequate for ensuring that degradation that could impact the battery's ability to perform its intended function has not occurred. Therefore, these changes are considered acceptable.

L.7 The battery cell electrolyte temperature limit of CTS 4.8.2.3.2.b.3 (ITS SR 3.8.6.3) has been slightly decreased to allow electrolyte temperature to be equal to 60° F. The engineering design calculation for the 125V 1E batteries assumes that minimum battery cell electrolyte temperature is 60°F, not greater than 60°F. 1

RELOCATED SPECIFICATIONS

None

<CTS>

ACTIONS (continued)

Division 1, 2, or 3 — 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
5 F 2. Two ~~[required]~~ DGs inoperable. OR Division 2 DG and the required opposite unit Division 2 DG inoperable. — 6	F 5 2.1 Restore one 4 ~~[required]~~ DG to OPERABLE status.	2 hours OR 72 — 8 ~~24~~ hours if Division 3 DG is inoperable
~~F. One [required] [automatic load sequencer] inoperable.~~ 2	~~-------REVIEWER'S NOTE-------- This Condition may be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads following a loss of offsite power independent of, or coincident with, a Design Basis Event. ----------~~ ~~F.1 Restore [required] [automatic load sequencer] to OPERABLE status.~~	~~[12] hours~~
15 G. Required Action and Associated Completion Time of Condition A, 5 B, C, D, ~~[or]~~ E, 4 or F not met. — 2	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	12 hours 36 hours
H. Three or more 4 ~~[required]~~ AC sources inoperable.	H.1 Enter LCO 3.0.3.	Immediately

3.8.1.1 Actf
3.8.1.1 Acti
<DOC L.5>
3.8.1.1 Acta
3.8.1.1 Actb
3.8.1.1 Actc
3.8.1.1 Acte
3.8.1.1 Actf
3.8.1.2 Actc
<DOC A.10>

5 OR Required Action B.2, B.3, or B.4 and associated Completion Time not met.

Portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant ~~is~~ is maintained or enhanced [15]

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<4.8.1.1.2.d.11> SR 3.8.1.17 ----------NOTE---------- This Surveillance shall not (Normally) be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. [12] ----------	TSTF-283 [24] [4]
Verify, with a (required) [9] DG operating in test mode and connected to its bus, an actual or simulated ECCS initiation signal overrides the test mode by [4] [1] a. Returning DG to ready-to-load operation; and [1]	{~~18~~ 24 months} [4] (a. For Division 1 and 2 DGs) [D]
[2] ~~b. Automatically energizing the emergency load from offsite power.~~	
<4.8.1.1.2.d.12> SR 3.8.1.18 ----------NOTE---------- This Surveillance shall not (Normally) be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. [12] [4] ----------	TSTF-283 [4]
[1] (For Division 1 and 2 DGs Only) Verify interval between each sequenced load block is ~~within ± 10% of design interval~~ (the) ~~for each load sequencer timer~~ [4] (time delay relay) (≥90%) [21]	{~~18~~ 24 months} [4]

(continued)

this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced.

b. For Division 3 DG, an actual or simulated DG overcurrent trip signal automatically disconnects the offsite power source while the DG continues to supply normal loads. [1]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
⟨4.8.1.1.2.d.6⟩ SR 3.8.1.19 ----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not (Normally) be performed in MODE 1, 2, or 3. [12] However, credit may be taken for unplanned events that satisfy this SR. ---------- Portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant (10) (is) [15] maintained or enhanced.	TSTF -283 (C) [4]
Verify, on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal:	(24) [18] months
a. De-energization of emergency buses;	
b. Load shedding from emergency buses; and (for Divisions 1 and 2 only) [1]	
c. DG auto-starts from standby condition and:	
1. energizes permanently connected loads in ≤ [10] seconds, (13) [4]	
2. energizes auto-connected emergency loads [4] (including) through ~~[load sequencer]~~, (time delay relays, where applicable) [4]	(C)
3. (achieves) steady state voltage ≥ [3744] (4010) V and ≤ [4576] (4310) V, [4]	(D)
4. (achieves) steady state frequency ≥ {58.8} Hz and ≤ {61.2} Hz, and [16] (maintains)	
5. supplies permanently connected and auto-connected emergency loads for ≥ {5} minutes.	

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
<4.8.1.1.2.e> SR 3.8.1.20	----------NOTE---------- All DG starts may be preceded by an engine prelube period. ----------	
	Verify, when started simultaneously from standby condition, ~~{each}~~ ~~([Division 1, 2, and 3])~~ DG achieves, in ≤ ~~([10])~~ seconds, voltage ≥ ~~([3744])~~ V and ~~(≤ [4576] V and~~ frequency ≥ ~~{58.8}~~ Hz ~~(and ≤ [61.2] Hz)~~.	10 years

[17] TSTF 163 changes not adopted

[9] required

4010

(13)

[4]

[D]

Insert SR 3.8.1.21 [3]

<CTS>

< 3.8.1.2 Action d >

Insert ACTION D

D.	Required offsite circuit or DG of LCO Item d. inoperable.	D.1	Declare associated standby gas treatment subsystem, control room area filtration subsystem, and control room area ventilation air conditioning subsystem inoperable.	Immediately

<CTS>

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

<DOC A.2> LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

<DOC A.2> APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

<DOC L.1>

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

<DOC L.1>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with stored fuel oil level: 1. For [DG 11 or 12], < [62,000] gal and ≥ [49,000] gal; and (31,000 / 26,550) 2. For [DG 13], < [41,200] gal and ≥ [33,500] gal. (29,750 / 25,550)	A.1 Restore fuel oil level to within limits. (In the fuel oil storage tank for the Division 1 and Division 2 DGs, and the opposite unit Division 2 DG; In the combined day tank and fuel oil storage tank for the Division 3 DG)	48 hours
~~B. One or more DGs with lube oil inventory: 1. For [DG 11 or 12], < [ ] gal and ≥ [425] gal; and 2. For [DG 13], < [ ] gal and ≥ [ ] gal.~~	~~B.1 Restore lube oil inventory to within limits.~~	~~48 hours~~

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.3.1 <LCO 3.8.1.1.b.1.b> <LCO 3.8.1.1.b.2> <LCO 3.8.1.2.b.2> <4.8.1.1.2.a.1> <4.8.1.1.2.a.2> <4.8.1.2>	Verify each fuel oil storage tank contains: a. ≥ [62,000] gal of fuel for [DGs 11 and 12;] and b. ≥ [41,200] gal of fuel for [DG 13]. (≥ 31,000 gal of fuel in each fuel oil storage tank for the Division 1 and Division 2 DGs and the opposite unit Division 2 DG. ≥ 28,750 gal of fuel in the combined fuel oil storage tank and day tank for the Division 3 DG.)	31 days (4) (C)
SR 3.8.3.2	Verify lube oil inventory is: a. ≥ [ ] gal for [DGs 11 and 12;] and b. ≥ [ ] gal for [DG 13].	31 days (6) (4) (1)
SR 3.8.3.3 (2) <DOC A.4> <4.8.1.2>	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program (1)
SR 3.8.3.4 (3) <4.8.1.1.2.a.7> <4.8.1.2>	Verify each DG air start receiver pressure is ≥ [225] (200) psig.	31 days (1) (C)
SR 3.8.3.5 (4) (1)	Check for and remove accumulated water from each fuel oil storage tank.	[31] (92) days (4) (C)
SR 3.8.3.6	For each fuel oil storage tank: a. Drain the fuel oil; b. Remove the sediment; and c. Clean the tank.	10 years (TSTF-2)

Insert LCO

a. One Division 1 125 VDC or Division 2 125 VDC electrical power subsystem capable of supplying one division of the onsite Class 1E DC Electrical Power Distribution System required by LCO 3.8.8, "Distribution Systems – Shutdown"; C D

b. The Division 3 125 VDC electrical power subsystem, when the Division 3 onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.8; and D

c. The opposite unit Division 2 125 VDC electrical power distribution subsystem, when the opposite unit Division 2 onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.8.

Insert ACTION A

A.	-------NOTE------- Not applicable when the opposite unit is in MODE 1, 2, or 3. ---------- One or more required Division 1, 2, and 3 DC electrical power subsystems inoperable.	A.1	Verify associated DC electrical power distribution subsystem is energized by OPERABLE opposite unit DC electrical power subsystem.	1 hour C
		AND		
		A.2	Restore required Division 1, 2, and 3 DC electrical power subsystem to OPERABLE status.	72 hours C

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.8.5 - DC SOURCES—SHUTDOWN

1. The proper LCO number has been provided. This change was necessary due to the deletion of ISTS 3.8.7, "Inverters—Operating," and ISTS 3.8.8, "Inverters—Shutdown." Also, an additional SR was added to be consistent with changes made to ITS 3.8.4.

2. The brackets have been removed and the proper plant specific information/value has been provided.

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. This clarification is necessary because defaulting to LCO 3.0.3 during irradiated fuel assembly movement in Mode 1, 2, or 3 would require the reactor to be shutdown, but would not require suspension of movement of irradiated fuel assemblies. Therefore, the proposed Note ensures that proper actions are taken when moving irradiated fuel assemblies in Mode 1, 2, or 3 (i.e., LCO 3.0.3 is not applicable and cannot be used in lieu of suspending fuel movement as required by the ACTIONS of the LCO). This change is also consistent with TSTF-36, Rev. 4 and the CTS.

4. The design of the LaSalle 1 and 2 DC Electrical Power System provides cross-ties between Unit 1 and Unit 2, such that a divisional DC source on one unit can provide power to the same DC distribution division on the opposite unit. Therefore, a new ACTION has been provided, consistent with current licensing basis as modified by the Discussion of Changes for ITS 3.8.5. The following ACTION has been renumbered, to reflect this addition. In addition, changes have also been made to ISTS 3.8.5 Condition A (see new second Condition) due to opposite unit DC source requirements. This is also consistent with current licensing basis.

5. Editorial change made to match the words in the LCO and ACTION requirements.

6. Change made to be consistent with the Writers Guide.

7. The ISTS LCO, as modified by TSTF-204, is not specific as to what the Division 1 or 2 source must be powering. The LCO has been modified to require the division to be powering a DC division required OPERABLE by LCO 3.8.8. D

<CTS>

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

and opposite unit Division 2 [1] D

<DOC A.2>
<DOC A.6>

LCO 3.8.6 Battery cell parameters for the {Division 1, 2, and 3} batteries shall be within the limits of Table 3.8.6-1. TSTF-278

<DOC A.6>

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

<DOC A.4>

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
<Table 4.8.2.3.2-1 fnote (1) Table 4.8.2.3.2-1 fnote (2)> <DOC L.5> A. One or more batteries with one or more battery cell parameters not within Category A or B limits. Table 3.8.6-1 TSTF-278	A.1 Verify pilot cell{s} electrolyte level and float voltage meet Table 3.8.6-1 Category C limits. [1]	1 hour
	AND	
	A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours AND Once per 7 days thereafter
	AND	
	A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days

(continued)

BASES

Insert Applicability -1 [5]

APPLICABILITY (continued)

entered, the Division 3 AC sources provide no additional assurance of meeting the above criteria.

[5] Insert Applicability-2

AC power requirements for MODES 4 and 5 and other conditions in which AC sources are required are covered in LCO 3.8.2, "AC Sources—Shutdown." [8]

ACTIONS

A.1

To ensure a highly reliable power source remains, it is necessary to verify the availability of the remaining required offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in the Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition D, for two required offsite circuits inoperable, is entered. [5] [12]

A.2

Required Action A.2, which only applies if the division cannot be powered from an offsite source, is intended to provide assurance that an event with a coincident single failure of the associated DG does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions (i.e., single division systems are not included, although, for this Required Action, Division 3 (HPCS System) is considered redundant to Division 1 and 2 ~~Emergency Core Cooling Systems~~ (ECCS). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has no offsite power available. [1] [4]

The Completion Time for Required Action A.2 is intended to allow time for the operator to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The division has no offsite power available supplying its loads; and [4]

(continued)

Insert Applicability-1

In addition, when this Note allowance is being used, both AC sources could be inoperable such that the Division 3 AC distribution subsystem is de-energized. In this case (the Division 3 AC electrical power distribution subsystem inoperable), LCO 3.0.6 would not preclude entry into the Distribution System ACTIONS since, with the Division 3 AC sources not required OPERABLE as allowed by this Note, the Division 3 AC sources cannot be considered as a support system to the Division 3 AC distribution subsystem. Thus, as required by LCO 3.0.2, the Distribution System-Operating ACTIONS for the inoperable Division 3 AC electrical power distribution subsystem must be entered.

D

Insert Applicability-2

D

Note 2 has been added taking exception to the Applicability requirements for the required opposite unit's Division 2 DG in LCO 3.8.1.c, provided the associated required equipment is inoperable (i.e., one SGT subsystem, one primary containment hydrogen recombiner subsystem, one control room area filtration subsystem, and one control room area ventilation air conditioning subsystem). This exception is intended to allow declaring the opposite unit's Division 2 supported equipment inoperable either in lieu of declaring the opposite unit's Division 2 DG inoperable, or at any time subsequent to entering ACTIONS for an inoperable opposite unit Division 2 DG. This exception is acceptable since, with the opposite unit powered Division 2 equipment inoperable and the associated ACTIONS entered, the opposite unit Division 2 DG provides no additional assurance of meeting the above criteria.

(The test can consist of a single rate if the rate employed for the performance discharge test exceeds the one minute rate and continues to envelope the duty cycle of the service test.)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.7 (continued)

9

The modified performance discharge test is a simulated duty cycle, consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

1 normally

discharge

1 when the modified performance discharge test is performed in lieu of a service test

performance 4

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

9

5 125 V

TSTF-283

INSERT SR 3.8.4.7-1

The reason for Note 2 is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy the Surveillance.

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

9 A battery modified performance discharge test is described in the Bases for SR 3.8.4.7. Either the battery performance discharge test or the modified performance discharge test is

(continued)

BASES

REFERENCES (continued)

5. FSAR, Chapter 6.
6. FSAR, Chapter 15.
7. Regulatory Guide 1.93, December 1974.
8. IEEE Standard 450, 1987.
9. Regulatory Guide 1.32, February 1977.
10. Regulatory Guide 1.129, December 1974.
11. IEEE Standard 485.

BASES

LCO (continued)

subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. ~~OPERABLE vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated [inverter via inverted DC voltage, inverter using internal AC source, or Class 1E constant voltage transformer].~~

Insert B3.8.7 LCO

In addition, at least one tie breaker~~s~~ between the redundant Division 2, safety related ~~AC~~, DC, ~~and AC vital bus~~ power distribution subsystems, ~~if they exist,~~ must be open. This prevents any emergency electrical malfunction in ~~any~~ one power distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s). If ~~any~~ tie breaker~~s~~ ~~are closed, the affected redundant~~ is not open, then both Division 2 DC electrical power distribution subsystems are considered inoperable. ~~This~~ The restriction of maintaining electrical separation applies to the onsite, safety related, redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 4.16 kV Source buses from being powered from the same offsite ~~circuit~~.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained, in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 4 and 5 and other conditions in which AC and DC electrical power distribution subsystems are required are covered in the Bases for LCO 3.8.~~10~~ 8, "Distribution Systems—Shutdown."

ACTIONS

A.1

With one or more Division 1 ~~or~~ and 2 required AC buses, load centers, motor control centers, or distribution panels ~~(except AC vital buses), in one division~~ inoperable, and a loss of function has not yet occurred, the

(continued)

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.8.7 - DISTRIBUTION SYSTEMS — OPERATING

1. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description or licensing basis description.

3. The brackets have been removed and the proper plant specific information/valve has been provided.

4. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

5. This change has been made since Section 3.5, "ECCS and RCIC System," provides the appropriate limits that are affected by the systems in this LCO.

6. Typographical/grammatical error corrected.

7. This change has made to be consistent with the Applicability of LCO 3.8.8.

8. The proper LCO number has been used.

9. Changes have been made to match the Specification.

1D

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.9.1 - REFUELING EQUIPMENT INTERLOCKS

1. The current wording of ISTS 3.9.1 and the associated Applicability could imply that all the refueling equipment interlocks are required at all times during in-vessel fuel movement. The Current Licensing Basis only requires the interlocks associated with the refuel position, not those associated with other positions of the reactor mode switch, and only when the reactor mode switch is in the refuel position, not when it is in the shutdown position. Therefore, to avoid confusion, the LCO and Applicability have been modified to specifically state that the refueling interlocks are those associated with the refuel position, and that it is applicable when the reactor mode switch is in the refuel position. This change is also consistent with TSTF-232.

2. The current licensing basis of LaSalle 1 and 2 refueling equipment interlocks have been provided.

3. Changes have been made consistent with proposed TSTF-225, Revision 1. 1 C 1 D

3.10 SPECIAL OPERATIONS

3.10.5 Multiple Control Rod Withdrawal – Refueling

LCO 3.10.5 The requirements of LCO 3.9.4, "Control Rod Position Indication"; and LCO 3.9.5, "Control Rod OPERABILITY – Refueling," may be suspended, and the "full-in" position indicators may be bypassed for any number of control rods in MODE 5, to allow withdrawal of these control rods, removal of associated control rod drives (CRDs), or both, provided the following requirements are met:

a. The four fuel assemblies are removed from the core cells associated with each control rod or CRD to be removed;

b. All other control rods in core cells containing one or more fuel assemblies are fully inserted; and

c. Fuel assemblies shall not be loaded into or shuffled within the reactor pressure vessel. 1 D

APPLICABILITY: MODE 5 with LCO 3.9.4 or LCO 3.9.5 not met.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the above requirements not met.	A.1 Suspend withdrawal of control rods and removal of associated CRDs.	Immediately
	AND	
	A.2.1 Initiate action to fully insert all control rods in core cells containing one or more fuel assemblies.	Immediately
	OR	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Initiate action to satisfy the requirements of this LCO.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.5.1	Verify the four fuel assemblies are removed from core cells associated with each control rod or CRD removed.	24 hours
SR 3.10.5.2	Verify all other control rods in core cells containing one or more fuel assemblies are fully inserted.	24 hours
SR 3.10.5.3	Verify fuel assemblies are not being loaded into or shuffled within the reactor pressure vessel.	24 hours

10

BASES

APPLICABLE SAFETY ANALYSES (continued)

withdrawn control rod if all fuel has been removed from the cell. With no fuel assemblies in the core cell, the associated control rod has no reactivity control function and is not required to remain inserted. Prior to reloading fuel into the cell, however, the associated control rod must be inserted to ensure that an inadvertent criticality does not occur, as evaluated in the Reference 1 analysis.

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

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Operation in MODE 5 with LCO 3.9.4, "Control Rod Position Indication," or LCO 3.9.5, "Control Rod OPERABILITY – Refueling," not met, can be performed in accordance with the Required Actions of these LCOs without meeting this Special Operations LCO or its ACTIONS. If multiple control rod withdrawal or removal, or CRD removal is desired, all four fuel assemblies are required to be removed from the associated cells. Prior to entering this LCO, any fuel remaining in a cell whose CRD was previously removed under the provisions of another LCO must be removed. "Withdrawal" in this application includes the actual withdrawal of the control rod as well as maintaining the control rod in a position other than the full-in position, and reinserting the control rod.

Loading of fuel assemblies into or shuffling within the reactor pressure vessel is prohibited when multiple control rods are withdrawn. This restriction is consistent with existing conditions to the facility operating licenses. 1(D)

APPLICABILITY

Operation in MODE 5 is controlled by existing LCOs. The exceptions from other LCO requirements (e.g., the ACTIONS of LCO 3.9.4 or LCO 3.9.5) allowed by this Special Operations LCO are appropriately controlled by requiring all fuel to be removed from cells whose "full-in" indicators are allowed to be bypassed.

(continued)

ADMINISTRATIVE (continued)

A.6 An alternative Required Action (ITS 3.10.5 Required Action A.3.1) has been added to the CTS 3.9.10.2 Action to initiate action to fully insert all control rods immediately, in lieu of meeting the requirements of the LCO. Since this new Required Action results in effectively exiting this Special Operations LCO and restores operation consistent with normal requirements for failure to meet the LCOs which were suspended by the Special Operations LCO (i.e., all control rods inserted), it is administrative (since use of the Special Operations LCOs are optional as described in proposed LCO 3.0.7).

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 A restriction on fuel assembly loading into or shuffing within the reactor pressure vessel with control rods withdrawn has been provided in ITS 3.10.5.c, consistent with existing conditions of the Operating Licenses (License Conditions 2.C.(37) and 2.C.(21) for Units 1 and 2, respectively). This will help ensure a reactivity excursion cannot occur with the requirements of this LCO not met. A new Surveillance Requirement has also been added (proposed SR 3.10.5.3) to verify, every 24 hours, fuel assemblies are not being loaded into or shuffled within the reactor pressure vessel. The addition of SR 3.10.5.3 represents an additional restriction on plant operation. 1D 1D 1D

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

"Specific"

L.1 The requirement in CTS 3.9.10.2.a and CTS 4.9.10.2.1.a to "lock" the reactor mode switch in Shutdown or Refuel and the explicit requirement for the reactor mode switch to be OPERABLE is proposed to be deleted. Reactor mode switch OPERABILITY is included as part of the OPERABILITY of the required interlocks and control rod blocks. Furthermore, the position of the reactor mode switch is adequately controlled by the MODES definition Table (ITS Table 1.1-1). Reactor mode switch positions other than Refuel and Shutdown result in the unit entering some other MODE; with the associated Technical Specification compliance requirements of that MODE and of proposed LCO 3.0.1. 1B

<CTS>

3.10 SPECIAL OPERATIONS

3.10.8 Multiple Control Rod Withdrawal—Refueling

<LCO 3.9.10.2> LCO 3.10.8 <DOC M.1>

The requirements of ~~LCO 3.9.3, "Control Rod Position";~~ LCO 3.9.4, "Control Rod Position Indication"; and LCO 3.9.5, "Control Rod OPERABILITY—Refueling," may be suspended, and the "full-in" position indicators may be bypassed for any number of control rods in MODE 5, to allow withdrawal of these control rods, removal of associated control rod drives (CRDs), or both, provided the following requirements are met:

a. The four fuel assemblies are removed from the core cells associated with each control rod or CRD to be removed;

b. All other control rods in core cells containing one or more fuel assemblies are fully inserted; and

c. ~~Fuel assemblies shall only be loaded in compliance with an approved [spiral] reload sequence.~~ Fuel assemblies shall not be loaded into or shuffled within the reactor pressure vessel.

<Appl 3.9.10.2> APPLICABILITY: MODE 5 with ~~LCO 3.9.3,~~ LCO 3.9.4 or LCO 3.9.5 not met.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.9.10.2 Act> A. One or more of the above requirements not met.	A.1 Suspend withdrawal of control rods and removal of associated CRDs.	Immediately
	~~AND~~ ~~A.2 Suspend loading fuel assemblies.~~	~~Immediately~~
	AND	(continued)

⟨CTS⟩

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.9.10.2 Act⟩ A. (continued)	A.6.1 Initiate action to fully insert all control rods in core cells containing one or more fuel assemblies.	Immediately
	OR	
	A.6.2 Initiate action to satisfy the requirements of this LCO.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
⟨4.9.10.2.1 4.9.10.2.1.e⟩ SR 3.10.6.1	Verify the four fuel assemblies are removed from core cells associated with each control rod or CRD removed.	24 hours
⟨4.9.10.2.1 4.9.10.2.1.d⟩ SR 3.10.6.2	Verify all other control rods in core cells containing one or more fuel assemblies are fully inserted.	24 hours
⟨DOC M.1⟩ SR 3.10.6.3	~~NOTE Only required to be met during fuel loading.~~	
	~~Verify fuel assemblies being loaded are in compliance with an approved [spiral] reload sequence.~~ are not being loaded into or shuffled within the reactor pressure vessel.	24 hours

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.10.5 - MULTIPLE CONTROL ROD WITHDRAWAL — REFUELING

1. Typographical/grammatical error corrected.

2. ISTS 3.10.5 permits movement (loading or unloading) of fuel assemblies within the reactor pressure vessel in MODE 5 with more than one control rod not full-in provided fuel loading is performed in compliance with an approved spiral reload sequence, and other core configurations are met. The facility operating licenses for LaSalle 1 and 2 each contain a condition that prohibits loading of fuel assemblies into or shuffling within the reactor pressure vessel unless all control rods, except one, are fully inserted during refueling in MODE 5. The changes to ISTS LCO 3.10.5, Required Action A.2, and SR 3.10.5.3 reflect these license conditions, and will prohibit the loading of fuel assemblies into or shuffling within the reactor pressure vessel when multiple control rods are not fully inserted. 1D 1D

3. ISTS 3.10.6 is renumbered as ITS 3.10.5 as a result of the deletion of ISTS 3.10.1, "Inservice Leak and Hydrostatic Testing Operation."

BASES

APPLICABLE SAFETY ANALYSES (continued)

the cell. With no fuel assemblies in the core cell, the associated control rod has no reactivity control function and is not required to remain inserted. Prior to reloading fuel into the cell, however, the associated control rod must be inserted to ensure that an inadvertent criticality does not occur, as evaluated in the Reference 1 analysis.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Operation in MODE 5 with LCO 3.9.3, "Control Rod Position," LCO 3.9.4, "Control Rod Position Indication," or LCO 3.9.5, "Control Rod OPERABILITY—Refueling," not met, can be performed in accordance with the Required Actions of these LCOs without meeting this Special Operations LCO or its ACTIONS. If multiple control rod withdrawal or removal, or CRD removal is desired, all four fuel assemblies are required to be removed from the associated cells. Prior to entering this LCO, any fuel remaining in a cell whose CRD was previously removed under the provisions of another LCO must be removed. "Withdrawal" in this application includes the actual withdrawal of the control rod as well as maintaining the control rod in a position other than the full-in position, and reinserting the control rod.

When loading fuel into the core with multiple control rods withdrawn, special spiral reload sequences are used to ensure that reactivity additions are minimized. Spiral reloading encompasses reloading a cell (four fuel locations immediately adjacent to a control rod) on the edge of a continuous fueled region (the cell can be loaded in any sequence). Otherwise, all control rods must be fully inserted before loading fuel.

Loading of fuel assemblies into and shuffling within the reactor pressure vessel is prohibited when multiple control rods are withdrawn. This restriction is consistent with existing conditions to the facility operating licenses.

(continued)