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December 18, 2000

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 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

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, "Response to Request for Additional Information," dated September 22, 2000.

5) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Response to Request for Additional Information," dated October 5, 2000.

6) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Response to Request for Additional Information," dated October 9, 2000.

7) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Response to Request for Additional Information," dated November 20, 2000.

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 and 3 subsequently supplemented the proposed amendment.

This letter submits further changes for the proposed amendment as Attachments 1, 2, and 3 to this letter. Each change and its source is described in the "Summary of Changes" section of Attachments 1, 2, and 3. The sources include the incorporation of recent TS amendments, the incorporation of newly approved Technical Specification Task Force Travelers, changes made based upon discussions in meetings and telephone conferences between Commonwealth Edison personnel and the staff, changes committed to in References 4, 5, 6, and 7, and minor corrections to various sections of the proposed amendment.

Concerning Reference 1, Attachment B, Section III.B stated the following. "For LaSalle County Station, Units 1 and 2, some but not all of the Rosemount transmitters have been replaced. LaSalle County Station has committed, in response to NRC Bulletin No. 90-01 and Supplement 1, to an enhanced monitoring program for these transmitters. This commitment will be addressed in separate correspondence." The LaSalle County Station response to NRC Bulletin 90-01, Supplement 1, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount," dated December 22, 1992, committed to maintain an enhanced surveillance program until the transmitters have reached maturity. The twenty-five Category 1.c transmitters and thirty-four Category 1.d transmitters have reached maturity as defined in the NRC Bulletin 90-01, Supplement 1 (>60,000 psi-months). These transmitters have been reclassified as Category 1.e. Therefore, this commitment is satisfied.

The proposed changes 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.

ComEd is notifying the State of Illinois of these 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 C to Dresden Improved Technical Specifications Document
Attachment 2 - Revision C to LaSalle Improved Technical Specifications Document
Attachment 3 - Revision C to Quad Cities Improved Technical Specifications Document

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 C 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 18th day of December, 2000.

" OFFICIAL SEAL "
Joseph V. Sipek
Notary Public, State of Illinois
My Commission Expires 11/24/2001

Notary Public

ATTACHMENT 1

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

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

This attachment provides a brief summary of the changes in Revision C 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 and B submitted to the NRC by letters dated June 5, 2000 and September 1, 2000, respectively.

Changes committed to in the ComEd Request For Additional Information (RAI) responses for Sections 3.2, 3.3, 3.4, 3.6, 3.7, 3.8, and 3.9, and Chapter 5.0 are provided in this Revision C submittal. In addition, changes to various Sections/Chapters of the entire submittal have been made based on discussions with the NRC reviewers, approved Technical Specification Task Force (TSTF) changes, and editorial corrections.

Section 3.2

1. The change committed to in the ComEd response to RAI 3.2.1-1 has been made. This change affects ITS 3.2.1 Bases page B 3.2.1-1 and the ISTS Bases markup page B 3.2-1.

2. A typographical error has been corrected (removal of a comma). This change affects ITS 3.2.1 BAses page B 3.2.1-1.

Section 3.3

1. The change committed to in the ComEd response to RAI 3.3.1.1-01 has been made. This change affects the CTS markup for ITS 3.3.1.1, page 12 of 17.

2. The change committed to in the ComEd response to RAI 3.3.1.1-03 has been made. This change affects ITS 3.3.1.1, page 3.3.1.1-1 and Bases page B 3.3.1.1-22, the ISTS markup page 3.3-1 and Bases insert page B 3.3-21.

3. The change committed to in the ComEd response to RAI 3.3.1.1-06 has been made. This change affects the Justification for Deviations to ITS 3.3.1.1, JFD 11 (pages 2 and 3).

4. The change committed to in the ComEd response to RAI 3.3.1.2-01 has been made. This change affects the Discussion of Changes for ITS 3.3.1.2, DOC M.2 (page 2).

5. The change committed to in the ComEd response to RAI 3.3.1.2-02 has been made. This change affects the CTS markup for ITS 3.3.1.2, page 2 of 3.

6. The change committed to in the ComEd response to RAI 3.3.2.1-03 has been made. This change affects the Discussion of Changes for ITS 3.3.2.1, DOC L.2 (page 6).

7. The change committed to in the ComEd response to RAI 3.3.2.2-01 has been made. This change affects the Justification for Deviations to ITS 3.3.2.2, JFD 4 (page 1).

8. The change committed to in the ComEd response to RAI 3.3.3.1-05 has been made. This change affects the Discussion of Changes for ITS 3.3.3.1, DOC L.5 (page 7).

9. The change committed to in the ComEd response to RAI 3.3.4.1-04 has been made. This change affects the Discussion of Changes for ITS 3.3.4.1, DOC L.3 (pages 9 and 10).

10. The change committed to in the ComEd response to RAI 3.3.5.1-03 has been made. This change affects the CTS markup for ITS 3.3.5.1, page 6 of 17, the Discussion of Changes for ITS 3.3.5.1, DOC A.13 (page 3) and DOC L.1 (page 12), and the No Significant Hazards Consideration, NSHC L.1 (page 1).

11. The change committed to in the ComEd response to RAI 3.3.5.1-11 has been made. This change affects the Justification for Deviations to ITS 3.3.5.1, JFD 10 (page 2).

12. The change committed to in the ComEd response to RAI 3.3.6.1-01 has been made. This change affects ITS 3.3.6.1, pages 3.3.6.1-4, 3.3.6.1-5, 3.3.6.1-6, and 3.3.6.1-7, and Bases pages B 3.3.6.1-25, B 3.3.6.1-26, and B 3.3.6.1-27, the CTS markup for ITS 3.3.6.1, pages 1 of 12, 10 of 12, 11 of 12, and 12 of 12, the Discussion of Changes for ITS 3.3.6.1, DOC A.8 (page 2), DOC LD.1 (pages 5 and 6), and DOC LE.1 (page 7), the ISTS markup, pages 3.3-55, 3.3-57, 3.3-58, insert page 3.3-58, 3.3-59, insert page 3.3-60, 3.3-61, and 3.3-62, the Justifications for Deviations to ITS 3.3.6.1, JFD 3 (page 1) and JFD 11 (page 2), and the ISTS Bases markup, pages B 3.3-181, B 3.3-182, and B 3.3-183.

13. The change committed to in the ComEd response to RAI 3.3.6.1-04 has been made. This change affects ITS 3.3.6.1, Bases pages B 3.3.6.1-4 and B 3.3.6.1-15, and the ISTS Bases markup pages insert page B 3.3-154 and B 3.3-168.

14. The change committed to in the ComEd response to LaSalle RAI 3.3.4.1-03 has been made for Dresden for consistency between the sites. This change affects ITS 3.3.1.1 Bases pages B 3.3.1.1-18 and B 3.3.1.1-19, the ISTS Bases markup pages B 3.3-18 and B 3.3-19, and the Justification for Deviations to ITS Bases 3.3.1.1, JFD 8 (page 1).

15. The change committed to based on discussions with the NRC has been made. An editorial change concerning SRM monitoring capability was made to be consistent with the NUREG wording. This change affects ITS 3.3.1.2 Bases page B 3.3.1.2-2 and the ISTS Bases markup page B 3.3-36.

16. The change committed to based on discussions with the NRC has been made. A clarification has been made in the Bases to explain the Notes for SR 3.3.2.1.2 and SR 3.3.1.2.3. This change affects ITS 3.3.2.1 Bases page B 3.3.2.1-10 and the ISTS Bases markup page B 3.3-52.

17. The change committed to based on discussions with the NRC has been made. A correction was made to the Bases description and Discussion of Changes to explain how the RWM functions when an individual control rod is bypassed (taken out of service). This change affects ITS 3.3.2.1 Bases page B 3.3.2.1-13, the Discussion of Changes for ITS 3.3.2.1, DOC M.6 (page 3), and the ISTS Bases markup insert page B 3.3-54.

18. The change committed to based on discussions with the NRC has been made. The change adds a discussion of the Surveillance Requirement Note, consistent with the NUREG wording. This change affects ITS 3.3.2.2, Bases pages B 3.3.2.2-5 and B 3.3.2.2-6, and the ISTS Bases markup page B 3.3-60.

19. The change committed to based on discussions with the NRC has been made. A clarification has been added to the Bases description of ITS 3.3.3.1 for the Drywell Pressure Function. This change affects ITS 3.3.3.1 Bases page B 3.3.3.1-5 and the ISTS Bases markup page B 3.3-66.

20. The change committed to based on discussions with the NRC has been made. A clarification has been made in the logic description for the Low Pressure Coolant Injection System. This change affects ITS 3.3.5.1 Bases page B 3.3.5.1-4 and the ISTS Bases markup insert page B 3.3-103.

21. The change committed to based on discussions with the NRC has been made. Typographical errors in the description of the Reactor Water Cleanup System Isolation and Residual Heat Removal Shutdown Cooling System Isolation logic were noted. These errors have been corrected. This change affects ITS 3.3.6.1 Bases pages B 3.3.6.1-4 and B 3.3.6.1-5 and the ISTS Bases markup insert page B 3.3-155.

22. The change committed to based on discussions with the NRC has been made. Typographical errors (missing period and word) have been found and corrected. This change affects ITS 3.3.6.1 Bases page B 3.3.6.1-9 and the ISTS Bases markup insert page B 3.3-159.

23. The change committed to based on discussions with the NRC has been made. A correction was made to the Bases description of the Control Room Emergency Ventilation System Instrumentation description. This change affects ITS 3.3.7.1 Bases, pages B 3.3.7.1-1 and B 3.3.7.1-2, and the ISTS Bases markup, pages B 3.3-207, insert page B 3.3-207, and insert page B 3.3-208.

24. The change committed to based on discussions with the NRC has been made. The change ensures the Applicable Safety Analyses in the Bases of ITS 3.3.8.2 matches the LCO requirements. This change affects ITS 3.3.8.2 Bases page B 3.3.8.2-2 and the ISTS Bases markup page B 3.3-228.

25. The change committed to based on discussions with the NRC have been made. An editorial change was made to be consistent with descriptions used in other places in the Bases. This change affects ITS 3.3.8.2 Bases page B 3.3.8.2-2 and the ISTS Bases markup page B 3.3-228.

26. TSTF-205 has been incorporated, as committed to in Reference 1. This change affects ITS 1.1, pages 1.1-1, 1.1-2, and 1.1-4, the CTS markup for ITS Chapter 1.0, pages 1 of 12, 3 of 12, and 4 of 12, the ISTS 1.1 markup, pages 1.1-1, 1.1-2, and 1.1-5, ITS 3.3.1.1, Bases pages B 3.3.1.1-29, B 3.3.1.1-32, and B 3.3.1.1-33, ITS 3.3.1.2 Bases page B 3.3.1.2-8, ITS 3.3.2.1, Bases pages B 3.3.2.1-9, B 3.3.2.1-11, and B 3.3.2.1-12, ITS 3.3.2.2, Bases pages B 3.3.2.2-6 and B 3.3.2.2-7, ITS 3.3.4.1 Bases page B 3.3.4.1-9, ITS 3.3.5.1 Bases page B 3.3.5.1-41, ITS 3.3.5.2, Bases pages B 3.3.5.2-4 and B 3.3.5.2-5, ITS 3.3.6.1 Bases page B 3.3.6.1-25, ITS 3.3.6.2 Bases page B 3.3.6.2-10, ITS 3.3.7.1 Bases page B 3.3.7.1-7, ITS 3.3.8.1 Bases page B 3.3.8.1-7, ITS 3.3.8.2 Bases page B 3.3.8.2-6, the ISTS 3.3.1.1 Bases markup pages B 3.3-27, insert page B 3.3-27, B 3.3-29, and insert page B 3.3-29, the ISTS 3.3.1.2 Bases markup pages B 3.3-41

26. (cont'd)

and insert page B 3.3-41, the ISTS 3.3.2.1 Bases markup pages B 3.3-51, insert page B 3.3-51, B 3.3-53, and insert page B 3.3-53, the ISTS 3.3.2.2 Bases markup pages B 3.3-61 and insert page B 3.3-61, the ISTS 3.3.4.1 Bases markup page B 3.3-87, the ISTS 3.3.4.2 Bases markup pages B 3.3-98 and insert page B 3.3-98, the ISTS 3.3.5.1 Bases markup pages B 3.3-136 and insert page B 3.3-136, the ISTS 3.3.5.2 Bases markup pages B 3.3-149 and insert page B 3.3-149, the ISTS 3.3.6.1 Bases markup pages B 3.3-181 and insert page B 3.3-181, the ISTS 3.3.6.2 Bases markup pages B 3.3-194 and insert page B 3.3-194, the ISTS 3.3.6.3 Bases markup page B 3.3-205, the ISTS 3.3.7.1 Bases markup pages B 3.3-217 and insert page B 3.3-217, the ISTS 3.3.8.1 Bases markup pages B 3.3-225 and insert page B 3.3-225, the ISTS 3.3.8.2 Bases markup pages B 3.3-232 and insert page B 3.3-232, ITS 3.4.5 Bases page B 3.4.5-4, the ISTS 3.4.5 Bases markup pages B 3.4-32 and insert page B 3.4-32, ITS 3.9.1, Bases pages B 3.9.1-4 and B 3.9.1-5, ITS 3.9.2 Bases page B 3.9.2-3, the ISTS 3.9.1 Bases markup pages B 3.9-4 and insert page B 3.9-4, the ISTS 3.9.2 Bases markup pages B 3.9-7 and insert page B 3.9-7, and the Justification for Deviations to ITS Bases 3.9.2, JFD 2 (page 1).

27. TSTF-231 has been incorporated. This change affects ITS 3.3.1.1 Bases page B 3.3.1.1-18 and the ISTS Bases markup page B 3.3-18.

28. A typographical error has been corrected (deleted a comma). This change affects ITS 3.3.6.1 Bases page B 3.3.6.1-2.

29. A change has been made to ensure a more restrictive change is described. This change affects the CTS markup page for ITS 3.3.8.1, pages 4 of 6 and 4a of 6, and the Discussion of Changes for ITS 3.3.8.1, DOC A.6 (page 2) and DOC M.3 (page 3).

<u>Section 3.4</u>

1. The change committed to in the ComEd response to RAI 3.4.1-01 has been made. This change affects the CTS markup for ITS 3.4.1, page 3 of 3, the Discussion of Changes for ITS 3.4.1, DOC M.2 (deleted from page 2) and DOC L.4 (page 5), and the No Significant Hazards Consideration for ITS 3.4.1, NSHC L.4 (page 4).

2. The change committed to in the ComEd response to RAI 3.4.3-01 has been made. This change affects the ISTS 3.4.3 markup page 3.4-6.

3. The change committed to in the ComEd response to RAI 3.4.7-03 and related changes for consistency (deletion of the word "required" from the LCO) have been made. This change affects ITS 3.4.7, page 3.4.7-1 and Bases pages B 3.4.7-2 and B 3.4.7-4, the ISTS markup pages 3.4-18 and 3.4-19, the Justification for Deviations to ITS 3.4.7, JFD 5 (page 1), and the ISTS Bases markup pages B 3.4-38 and B 3.4-40.

4. The change committed to in the ComEd response to RAI 3.4.8-02 and related changes for consistency (deletion of the word "required" from the LCO) have been made. These changes affect ITS 3.4.8, pages 3.4.8-1 and 3.4.8-2, and Bases pages B 3.4.8-2 and B 3.4.8-4, the ISTS markup pages 3.4-21 and 3.4-22, the Justification for Deviations to ITS 3.4.8, JFD 5 (page 1), and the ISTS Bases markup pages B 3.4-43, insert page B 3.4-43, and B 3.4-45.

5. The change committed to in the ComEd response to RAI 3.4.9-03 has been made. This change affects the CTS markup for ITS 3.4.9, page 2 of 6, and the Discussion of Changes for ITS 3.4.9, DOC A.10 (page 3).

6. Markup errors were noted in ITS 3.4.3. These errors have been corrected. These changes affect the CTS markup for ITS 3.4.3, page 3 of 4 and the Discussion of Changes for ITS 3.4.3, DOC LA.2 (pages 2 and 3).

7. Markup errors were noted in ITS 3.4.8 in that the CTS requirement to monitor "pressure" was not adopted in the ITS. This change has been corrected. This change affects ITS 3.4.8 page 3.4.8-2, the ISTS markup for 3.4.8, page 3.4-22, and the Justification for Deviations to ITS 3.4.8, JFD 6 (page 1).

Section 3.6

1. The change committed to in the ComEd response to RAI 3.6.1.7-06 has been made, except that the specific cases have been added, consistent with the ISTS, in lieu of only referencing the UFSAR. This change affects ITS 3.6.1.7, Bases pages B 3.6.1.7-2 and B 3.6.1.7-3, the ISTS Bases markup, pages B 3.6-43 and B 3.6-44, and the Justification for Deviations to ITS 3.6.1.7 Bases, JFD 3 (page 1).

2. The changes committed to during discussions with the NRC to resolve RAI 3.6.1.1-2 and RAI 3.6.1.1-3 have been made. These changes affect 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, page 3 of 3, the Discussion of Changes for ITS 3.6.1.1, DOC LD.1 (pages 2 and 3), DOC L.3 (page 4), and DOC L.4 (deleted from page 5), the ISTS markup page 3.6-2, the ISTS Bases markup insert page B 3.6-4 and page B 3.6-5, and the No Significant Hazards Consideration for ITS 3.6.1.1, NSHC L.4 (deleted).

3. The changes committed to during discussions with the NRC to resolve RAI 3.6.4.1-3 have been made. These changes affect ITS 3.6.4.1, Bases pages B 3.6.4.1-3 and B 3.6.4.1-4, ITS 3.6.4.2, Bases pages B 3.6.4.2-5 and B 3.6.4.2-6, ITS 3.6.4.3, Bases pages B 3.6.4.3-4 and B 3.6.4.3-5, the ISTS 3.6.4.1 Bases markup pages B 3.6-99 and B 3.6-100, the ISTS 3.6.4.2 Bases markup page B 3.6-106, and the ISTS 3.6.4.3 Bases markup pages B 3.6-112, insert page B 3.6-112, and B 3.6-113.

4. The change committed to during discussions with the NRC to resolve RAI 3.6.4.2-1 has been made. This change affects the CTS markup for ITS 3.6.4.2, page 3 of 3, and the Discussion of Changes for ITS 3.6.4.2, DOC M.2 (page 2).

Section 3.7

1. The change committed to in the ComEd response to RAI 3.7.2-03 has been made. This change affects ITS 3.7.2, page 3.7.2-2 and Bases page B 3.7.2-3, the CTS markup for ITS 3.7.2, page 1 of 1, the Discussion of Changes for ITS 3.7.2, DOC LA.3 (page 3) and DOC L.1 (page 4), the ISTS markup page 3.7-8, the Justification for Deviations to ITS 3.7.2, JFD 8 (page 2), the ISTS Bases markup page B 3.7-16, and the No Significant Hazards Consideration for ITS 3.7.2, NSHC L.1 (page 1).

2. The change committed to based on discussions with the NRC has been made. The description of the ITS 3.7.8 Required Action A.1 Note in the Bases has been revised to be consistent with the ISTS wording. This change affects ITS 3.7.8 Bases page B 3.7.8-2, the ISTS Bases markup page B 3.7-38, and the Justification for Deviations to ITS Bases 3.7.8, JFD 5 (deleted from page 1).

3. Editorial changes have been made to ITS 3.7.2 (deletion of the term "unit" and "required"). These changes affect ITS 3.7.2, pages 3.7.2-1 and 3.7.2-2 and Bases pages B 3.7.2-2 and B 3.7.2-3, the ISTS markup, insert page 3.7-7 and page 3.7-8, and Bases pages B 3.7-14 and B 3.7-16.

Section 3.8

1. The change committed to in the ComEd response to RAI 3.8.1-1 has been made. This change affects ITS 3.8.1 Bases page B 3.8.1-6 and the ISTS Bases markup insert page B 3.8-4.

2. The change committed to in the ComEd response to RAI 3.8.1-8 has been made. This change affects ITS 3.8.1 page 3.8.1-14, the CTS markup for ITS 3.8.1, page 6 of 9, the Discussion of Changes for ITS 3.8.1, DOC A.13 (page 3) and LA.3 (page 8), and the ISTS markup page 3.8-15.

3. The change committed to in the ComEd response to RAI 3.8.2-3 has been made. This change affects the CTS markup for ITS 3.8.2, page 1 of 2, the Discussion of Changes for ITS 3.8.2, DOC L.3 (pages 4 and 5), and the No Significant Hazards Consideration for ITS 3.8.2, NSHC L.3 (page 4).

4. The change committed to in the ComEd response to RAI 3.8.3-3 has been made. This change affects the CTS markup for ITS 3.8.3, page 5 of 6, the Discussion of Changes for ITS 3.8.3, DOC LA.1 (page 2) and DOC L.3 (deleted from page 3), and the No Significant Hazards Consideration for ITS 3.8.3, NSHC L.3 (deleted).

5. The change committed to in the ComEd response to RAI 3.8.4-1 has been made. This change affects ITS 3.8.4 Bases page B 3.8.4-4 and ISTS Bases markup page B 3.8-52.

6. The change committed to in the ComEd response to RAI 3.8.4-3 has been made. This change affects ITS page 3.8.4-5 and Bases pages B 3.8.4.1, B 3.8.4-2, B 3.8.4-8, B 3.8.4-10, B 3.8.4-11, and B 3.8.4-12, the Discussion of Changes for ITS 3.8.4, DOC LA.2 (page 3), and the ISTS markup, page 3.8-25 and Bases pages B 3.8-50, insert page B 3.8-50, insert page B 3.8-53b, insert page B 3.8-53c, insert page B 3.8-53d, and B 3.8-54.

7. The change committed to in the ComEd response to RAI 3.8.5-2 has been made. In addition, TSTF-204 has been incorporated. This change affects ITS 3.8.5, page 3.8.5-1 and Bases pages B 3.8.5-1, B 3.8.5-2, B 3.8.5-3, and B 3.8.5-4, 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), DOC LA.1 (page 2), and DOC L.2 (page 3), the ISTS 3.8.5 markup page 3.8-28 and Bases pages B 3.8-60, insert page B 3.8-60, B 3.8-61, insert page B 3.8-61, and B 3.8-62, the ISTS 3.8.8 markup page 3.8-36 and Bases pages B 3.8-75, B 3.8-76, and B 3.8-77, and the No Significant Hazards Consideration for ITS 3.8.5, NSHC L.2 (page 2).

8. The change committed to in the ComEd response to RAI 3.8.6-1 has been made. This change affects ITS 3.8.6, page 3.8.6-4 and Bases page B 3.8.6-5, the ISTS markup page 3.8-33, the Justification for Deviations to ITS 3.8.6, JFD 5 (page 1), and the ISTS Bases markup page B 3.8-67.

9. The change committed to in the ComEd response to RAI 3.8.7-1 has been made. This change affects ITS 3.8.7, pages 3.8.7-1, 3.8.7-2, and 3.8.7-3, and Bases pages B 3.8.7-6, B 3.8.7-7, B 3.8.7-8, B 3.8.7-9, and B 3.8.7-10, the CTS markup for ITS 3.8.7, pages 1 of 2 and 2 of 2, the Discussion of Changes for ITS 3.8.7, DOC M.1 (pages 1 and 2), DOC M.2 (page 2), DOC M.3 (page 2), and DOC L.1 (pages 3 and 4), the ISTS markup pages 3.8-38, insert page 3.8-38, and 3.8-39, the Justification for Deviations to ITS 3.8.7, JFD 3 (page 1), JFD 4 (deleted, but a new JFD 4 was added per another change), JFD 5 (page 2), and JFD 6 (page 2), and the ISTS Bases markup pages B 3.8-82, B 3.8-83, B 3.8-84, B 3.8-85, B 3.8-86, insert page B 3.8-86, and B 3.8-87.

10. The change committed to in the ComEd response to RAI 3.8.7-3 has been made. This change affects ITS 3.8.7 page 3.8.7-2 and Bases page B 3.8.7-10, the Discussion of Changes for ITS 3.8.7, DOC M.2 (page 2) and DOC L.1 (page 4), the ISTS markup page 3.8-39, the Justification for Deviations to ITS 3.8.7, JFD 4 (page 1), and the ISTS Bases markup page B 3.8-87.

11. The change committed to in the ComEd response to RAI 3.8.8-1 has been made. This change affects the CTS markup for ITS 3.8.8, page 2 of 3, the Discussion of Changes for ITS 3.8.8, DOC L.1 (page 3), and the No Significant hazards Consideration for ITS 3.8.8, NSHC L.1 (page 1).

12. The change committed to in the ComEd response to Quad Cities RAI 3.8.1-09 has been made for Dresden for consistency between the sites. In addition, a typographical error was corrected ("and" changed to "or" in the LCO Section of the Bases). This change affects ITS 3.8.1 Bases page B 3.8.1-5 and the ISTS Bases markup insert page B 3.8-4.

13. The change committed to in the ComEd response to Quad Cities RAI 3.8.1-16 has been made for Dresden for consistency between the sites. This change affects ITS 3.8.1 Bases page B 3.8.1-22 and the ISTS Bases markup page B 3.8-19.

14. The change committed to in the ComEd response to Quad Cities RAI 3.8.1-19 has been made for Dresden for consistency between the sites. This change affects ITS 3.8.1 Bases page B 3.8.1-29 and the ISTS Bases markup insert page B 3.8-28.

15. The change committed to in the ComEd response to LaSalle RAI 3.8.1-19 has been made for Dresden for consistency between the sites. In addition, this change was also committed to based on discussions with the NRC. This change affects ITS 3.8.1, page 3.8.1-11 and Bases page B 3.8.1-27, the CTS markup for ITS 3.8.1, page 6 of 9, the Discussion of Changes for ITS 3.8.1, DOC M.7 (page 6), the ISTS 3.8.1 markup page 3.8-11, the Justification for Deviations to ITS 3.8.1, JFD 19 (page 7), and the ISTS Bases markup page B 3.8-25.

16. The change committed to in the ComEd response to Quad Cities RAI 3.8.2-04 has been made for Dresden for consistency between the sites. This change affects ITS 3.8.2 Bases page B 3.8.2-5 and the ISTS Bases markup page B 3.8-38.

17. The change committed to in the ComEd response to Quad Cities RAI 3.8.3-04 has been made for Dresden for consistency between the sites. This change affects ITS 3.8.3 Bases page B 3.8.3-1 and the ISTS Bases markup page B 3.8-41.

18. The change committed to in the ComEd response to Quad Cities RAI 3.8.4-05 has been made for Dresden for consistency between the sites. This change affects ITS 3.8.4 Bases page B 3.8.4-3 and ISTS Bases markup page B 3.8-51.

19. The change committed to in the ComEd response to Quad Cities RAI 3.8.7-01 has been made for Dresden for consistency between the sites. This change affects ITS 3.8.7 Bases page B 3.8.7-5, the ISTS Bases markup page B 3.8-81, and the Justification for Deviations to ITS Bases 3.8.7, JFD 10 (page 1).

20. The change committed to in the ComEd response to Quad Cities RAI 3.8.7-02 has been made for Dresden for consistency between the sites. This change affects ITS 3.8.7, Bases pages B 3.8.7-2, B 3.8.7-3, B 3.8.7-4, B 3.8.7-12, and B 3.8.7-13, and the ISTS Bases markup, pages B 3.8-79, B 3.8-80, insert page B 3.8-81, B 3.8-88, insert page B 3.8-88a, and insert page B 3.8-88b.

21. The change committed to based on discussions with the NRC has been made. The change was made to revise the DG load block Surveillance test (SR 3.8.1.18) criteria from a tolerance of $\pm$10% of the design interval to $\geq$ 90% of the design interval. This change affects ITS 3.8.1, page 3.8.1-14 and Bases page B 3.8.1-31, the CTS markup for ITS 3.8.1, page 8 of 9, the Discussion of Changes for ITS 3.8.1, DOC L.14 (pages 18 and 19), the ISTS markup page 3.8-15, the Justification for Deviations to ITS 3.8.1, JFD 14 (page 6), the ISTS Bases markup pages B 3.8-30 and insert page B 3.8-30, and the No Significant Hazards Consideration for ITS 3.8.1, NSHC L.14 (page 18).

22. The change committed to based on discussions with the NRC has been made. For consistency, the change was made to delete the word "required" from the ITS 3.8.1 ACTION D Note and add the word "required" to SR 3.8.1.1, including appropriate Bases changes, and add the word "required" to the ITS 3.8.2 ACTIONS Bases. This change affects ITS 3.8.1, pages 3.8.1-4 and 3.8.1-6, and Bases page B 3.8.1-15, ITS 3.8.2 Bases page B 3.8.2-5, the ISTS 3.8.1 markup, pages 3.8-4 and 3.8-6, and Bases page B 3.8-13, and the ISTS 3.8.2 Bases markup page B 3.8-38.

23. The change committed to based on discussions with the NRC has been made. The change was made to clarify how the performance of SR 3.8.1.2, SR 3.8.1.3, and SR 3.8.1.8 should be alternated between the two units. This change affects ITS 3.8.1, Bases pages B 3.8.1-19 and B 3.8.1-21, and the ISTS Bases markup insert page B 3.8-17 and insert page B 3.8-18.

24. The change committed to based on discussions with the NRC has been made. The term "shared" has been changed to "common" in the ITS 3.8.1 Bases. This change affects ITS 3.8.1 Bases page B 3.8.1-5 and the ISTS Bases markup page B 3.8-4.

25. The change committed to based on discussions with the NRC has been made. The change was made to clarify details in the Bases concerning the battery modified performance discharge test. This change affects ITS 3.8.4 Bases page B 3.8.4-15 and the ISTS Bases markup page B 3.8-57.

26. The change committed to based on discussions with the NRC has been made. The change was made to a Justification for Deviations to be consistent with a generic change being reviewed by the NRC. This change affects the Justification for Deviations to the Bases of ITS Bases 3.8.6, JFD 4 (page 1).

27. The change committed to based on discussions with the NRC has been made. A Bases clarification concerning the instrument bus automatic bus transfer feature has been made. This change affects ITS 3.8.7, Bases pages B 3.8.7-1 and B 3.8.7-2, and the ISTS Bases markup insert page B 3.8-79.

28. The change committed to based on discussions with the NRC has been made. A Note has been added to ITS 3.8.7 Required Action C.1 to clarify that it may be necessary to cascade to the Conditions and Required Actions of ITS 3.8.1 if the inoperable portion of the opposite unit distribution system affects the alternate offsite circuit. This change affects ITS 3.8.7, page 3.8.7-2 and Bases page B 3.8.7-9, the Discussion of Changes for ITS 3.8.7, DOC M.3 (page 3), the ISTS markup, insert page 3.8-38 and Bases insert page B 3.8-86.

29. The change committed to based on discussions with the NRC has been made. A clarification was made to the ITS 3.8.7 Bases concerning the tie breakers between redundant safety related AC or DC power distribution systems. This change affects ITS 3.8.7 Bases page B 3.8.7-5 and the ISTS Bases markup page B 3.8-81.

30. The change committed to based on discussions with the NRC has been made. A clarification has been made to the Bases of ITS 3.8.8 LCO that the opposite unit Division 2 electrical distribution subsystem must also be energized. This change affects ITS 3.8.8 Bases page 3.8.8-2 and the ISTS Bases markup page B 3.8-90.

31. This change involves the changes proposed in a ComEd letter to the NRC (PSLTR# 00-0120) dated August 31, 2000, Request for Technical Specifications Change, Emergency Diesel Generator Surveillance Testing Requirements. The NRC project manager requested this change be technically justified in the ITS submittal in lieu of referencing the technical justification in the licensing amendment request. These changes affect the CTS markup for ITS 3.8.1, pages 2 of 9, 4 of 9, 5 of 9, and 6 of 9, and the Discussion of Changes for ITS 3.8.1, DOC A.13 (deleted from page 3) and DOC M.5 (pages 5 and 6).

32. A typographical error has been corrected in ITS 3.8.1 Bases (a period was inadvertently left out). This change affects ITS 3.8.1 Bases page B 3.8.1-7.

33. An error was noted in the SR 3.8.1.9 Bases concerning the transfer test between the normal and alternate qualified offsite circuits. This change has been made to match the actual SR requirement. This change affects ITS 3.8.1 Bases page B 3.8.1-23 and the ISTS Bases markup page B 3.8-20.

34. A typographical error has been corrected in ITS 3.8.2 (deleted the redundant term "ESS"). This change affects ITS 3.8.2 Bases page B 3.8.2-4.

35. The term "Division 1 or 2" has been added to ITS 3.8.4 Required Actions D.2, E.2, and F.1 for consistency with the associated Conditions. In addition, a typographical error was corrected in Condition H (the word "Associated" should

35. (cont'd)

not be capitalized). These changes affect ITS 3.8.4, pages 3.8.4-2, 3.8.4-3, and 3.8.4-4, the ISTS markup pages 3.8-24, insert page 3.8-24b, and insert page 3.8-24c, and the Justification for Deviations to ITS 3.8.4, JFD 7 (page 1).

36. Typographical errors have been corrected in ITS 3.8.5 (addition of word "the" and deletion of "(s)" from SR 3.8.5.1 Note and appropriate Bases, DOC, and JFD changes, correction of alignment problem, addition of word "of" to Applicable Safety Analyses Bases section). These changes affect ITS 3.8.5, page 3.8.5-2 and Bases pages B 3.8.5-2 and B 3.8.5-5, the Discussion of Changes for ITS 3.8.5, DOC L.1 (page 3), the ISTS markup page 3.8-29, the Justification for Deviations to ITS 3.8.5, JFD 4 (page 1), and the ISTS Bases markup page B 3.8-62.

Section 3.9

1. The change committed to in the ComEd response to RAI 3.9.1-2 has been made. This change affects ITS 3.9.1, page 3.9.1-1 and Bases page B 3.9.1-4, the Discussion of Changes for ITS 3.9.1, DOC L.3 (page 4), the ISTS markup, pages 3.9-1 and insert page 3.9-1, the Justification for Deviations to ITS 3.9.1, JFD 3 (page 1), the ISTS Bases markup pages B 3.9-3 and insert page B 3.9-3, and the No Significant Hazards Consideration for ITS 3.9.1, NSHC L.3 (pages 3 and 4).

Chapter 5.0

1. The change committed to in the ComEd response to RAI 5.0-1 has been made. This change affects the CTS markup for ITS 5.1, page 1 of 1, and the Discussion of Changes for ITS 5.1, DOC M.1 (page 1) and LA.2 (page 1).

2. The change committed to in the ComEd response to RAI 5.0-2 has been made. This change affects ITS 5.2 page 5.2-2, the CTS markup for ITS 5.2, page 3 of 3, the Discussion of Changes for ITS 5.2, DOC A.3 (deleted from page 1), DOC LA.1 (pages 1 and 2), DOC LA.2 (page 2), DOC LA.3 (page 2), and DOC L.1 (page 3), the ISTS markup pages 5.0-3 and 5.0-4, and the Justification for Deviations to ITS 5.2, JFD 8 (page 1).

3. The change committed to in the ComEd response to RAI 5.0-3 has been made. This change affects ITS 5.5 page 5.5-10 and ISTS markup page 5.0-15.

4. The change committed to in the ComEd response to RAI 5.0-4 has been made. This change affects ITS 5.5, pages 5.5-5 and 5.5-6, the CTS markup for ITS 5.5, pages 8 of 18 and 11 of 18, the Discussion of Changes for ITS 5.5, DOC A.11 (page 3), and the ISTS markup insert page 5.0-11.

5. The change committed to in the ComEd response to RAI 5.0-8 has been made. This change affects ITS 5.6, pages 5.6-3 and 5.6-4, the CTS markup for ITS 5.6, pages 3 of 5 and 4 of 5, the Discussion of Changes for ITS 5.6, DOC LA.2 (page 3), and the ISTS markup pages 5.0-20, insert page 5.0-20, and insert page 5.0-20a.

6. A typographical error was noted in that an incorrect ITS number was used in ITS 5.5. This change affects ITS 5.5, page 5.5-1.

7. A change has been made to ITS 5.5.8 to be consistent with the CTS. This change affects ITS 5.5 page 5.5-9 and the ISTS markup page 5.0-14.

8. A typographical error has been corrected (the first word of three sentences has been capitalized). This change affects ITS 5.5, pages 5.5-9 and 5.5-10, and the ISTS markup page 5.0-15.

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1.0 USE AND APPLICATION

1.1 Definitions

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

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	The APLHGR shall be applicable to a specific planar height and is equal to the sum of the LHGRs for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at the height.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

(continued)

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose

(continued)

1.1 Definitions (continued)

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWt.

(continued)

A.1 ITS Chapter 1.0

1.1 1.0 DEFINITIONS

Note to Definitions

~~The following terms are defined so that uniform interpretation of these specifications may be achieved.~~ The defined terms of this Section appear in capitalized type and ~~shall be~~ are applicable throughout these Technical Specifications and Bases.

A.1

ACTIONS

ACTIONS shall be that part of a Specification ~~which~~ that prescribes ~~remedial measures~~ Actions to be taken required under designated conditions within specified Completion Times.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

The ~~AVERAGE PLANAR LINEAR HEAT GENERATION RATE~~ APLHGR shall be applicable to a specific planar height and is equal to the sum of the ~~LINEAR HEAT GENERATION RATE(s)~~ LHGRs for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at the height.

~~CHANNEL~~

~~A CHANNEL shall be an arrangement of a sensor and associated components used to evaluate plant variables and generate a single protective action signal. A CHANNEL terminates and loses its identity where single action signals are combined in a TRIP SYSTEM or logic system.~~ A.2

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the CHANNEL output such that it responds within the necessary range and accuracy to known values of the parameter ~~which~~ that the CHANNEL monitors. The CHANNEL CALIBRATION shall encompass ~~the entire CHANNEL including the required sensor and alarm and/or trip functions, and shall include~~ all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total CHANNEL steps ~~such that the entire CHANNEL is calibrated~~. A.1 A.3

A.4 INSERT 1

CHANNEL CHECK

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

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A.1

ITS Chapter 1.0

1.1 **1.0 DEFINITIONS**

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be: A.1 A.1

a. Analog CHANNEL(s) - the injection of a simulated (or actual) signal into the CHANNEL as close to the sensor as practicable to verify OPERABILITY including required alarm and/or trip functions and CHANNEL failure trip. of all devices in the channel required for channel OPERABILITY A.3 C

b. Bistable CHANNEL(s) - the injection of a simulated signal into the sensor to verify OPERABILITY including required alarm and/or trip functions. A.3 L.1

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is tested. A.1 C

CORE ALTERATION

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

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

b. Control rod movement, provided there are no fuel assemblies in the associated control cell. core A.1

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

CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT (COLR) shall be is the unit specific document that provides core operating cycle specific parameter limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.9 5.6.5. Plant operation within these operating limits is addressed in individual specifications. A.1

CRITICAL POWER RATIO (CPR)

The CRITICAL POWER RATIO (CPR) shall be the ratio of is that power in the assembly which that is calculated by application of the applicable NRC approved critical power appropriate correlation(s) to cause some point in the assembly to experience transition boiling, divided by the actual assembly operating power. A.5 Insert into MCPR definition on page 1-4.

DOSE EQUIVALENT I-131

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

add the two additional thyroid dose conversion factor methods L.2

A.1

ITS Chapter 1.0

1.1 1.0 DEFINITIONS

~~FRACTION OF RATED THERMAL POWER (FRTP)~~
~~The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.~~ LA.1

~~FREQUENCY NOTATION~~
~~The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.~~ A.6

~~FUEL DESIGN LIMITING RATIO (FDLRX)~~
~~The FUEL DESIGN LIMITING RATIO (FDLRX) shall be the limit used to assure that the fuel operates within the end-of-life steady-state design criteria by, among other items, limiting the release of fission gas to the cladding plenum.~~ A.2

1.2 times the LHGR existing at a given location divided by the product of the transient LHGR and the fraction of RTP.

FUEL DESIGN LIMITING RATIO ~~for~~ CENTERLINE MELT (FDLRC)
The ~~FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC)~~ shall be ~~the limit used to assure that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of RATED THERMAL POWER.~~ A.1

A.7 a. Identified LEAKAGE

~~IDENTIFIED~~ LEAKAGE
~~IDENTIFIED~~ LEAKAGE shall be: ~~a)~~ 1. leakage into ~~primary containment collection systems~~ the drywell, such as that from pump seal or valve packing ~~leaks~~ that is captured and conducted to a sump or collecting tank, or ~~b)~~ 2. leakage into the ~~primary containment~~ drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of ~~the~~ leakage detection systems or not to be ~~PRESSURE BOUNDARY~~ LEAKAGE. A.8 A.1 A.7

~~LIMITING CONTROL ROD PATTERN (LCRP)~~
~~A LIMITING CONTROL ROD PATTERN (LCRP) shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.~~ A.2

LINEAR HEAT GENERATION RATE (LHGR)
The ~~LINEAR HEAT GENERATION RATE (LHGR)~~ shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. A.1

A.3 required for OPERABILITY

LOGIC SYSTEM FUNCTIONAL TEST ~~(LSFT)~~
A LOGIC SYSTEM FUNCTIONAL TEST ~~(LSFT)~~ shall be a test of all ~~required~~ logic components, ~~i.e., all required relays and contacts, trip units, solid state logic elements, etc.~~ of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested. A.3 A.1 C

Page 4 of 12

<CTS>

1.0 USE AND APPLICATION

<1.0>

1.1 Definitions

---------------------------------------NOTE---------------------------------------

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

Term	Definition
<A.1> ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
<A.1> AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	The APLHGR shall be applicable to a specific planar height and is equal to the sum of the [LHGRs] ~~(heat generation rate per unit length of fuel rod)~~ for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle [at the height]. 1
<A.4> CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass ~~the entire channel, including the required sensor, alarm, display, and trip functions, and shall include~~ all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps ~~so that the entire channel is calibrated~~. TSTF -205 C C
<A.1> CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or

(continued)

<CTS>

<1.0> 1.1 Definitions

CHANNEL CHECK (continued)

status derived from independent instrument channels measuring the same parameter.

<L.1> CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY ~~, including required alarm, interlock, display, and trip functions, and channel failure trips~~ of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps ~~so that the entire channel is tested~~.

TSTF-205

C

<A.1> CORE ALTERATION

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

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

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

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

<A.1> CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

<L.2> DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose

(continued)

<1.0> 1.1 Definitions (continued)

<A.1> LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all ~~required~~ logic components ~~(i.e., all required relays and contacts, trip units, solid state logic elements, etc.)~~ of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested. TSTF-205 C
required for OPERABILITY	
~~[MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)~~	~~The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.]~~ 3
<A.5> MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core [for each class of fuel]. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power. 1
<A.10> MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
<A.1> OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
<A.2> ~~PHYSICS TESTS~~	~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation.~~ 2

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the criteria specified in 10 CFR 50.46 are met during the postulated design basis loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) that determine the APLHGR limits are presented in References 1, 2, 3, and 4.

LOCA analyses are performed to ensure that the APLHGR limits are adequate to meet the peak cladding temperature (PCT) and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 1. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. A conservative multiplier is applied to the LHGR and APLHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR. APLHGR limits are typically set high enough such that the LHGR limits are more limiting than the APLHGR limits.

For single recirculation loop operation, an APLHGR limit multiplier is presented in the COLR. This additional limitation is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

The APLHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that ~~the fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the~~ criteria ~~limits~~ specified in 10 CFR 50.46 are met [1] [2] 1 C

APPLICABLE SAFETY ANALYSES

[2] ~~The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2.~~ The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), ~~anticipated operational transients, and normal~~ operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, ~~5, 6,~~ ~~and 7~~.

[1] ~~Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Refs. 5, 6, and 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.~~

~~Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at~~

(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 Function 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.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. <u>OR</u> A.2 Place associated trip system in trip.	12 hours 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. <u>OR</u> B.2 Place one trip system in trip.	6 hours 6 hours

(continued)

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation Function.

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

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.1.3	Calibrate the trip unit.	92 days
SR 3.3.6.1.4	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.6.1.5	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.6.1.6	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.1.7	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.6.1-1 (page 1 of 3)
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	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -56.77 inches
b. Main Steam Line Pressure − Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≥ 831 psig
c. Main Steam Line Pressure − Timer	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 0.280 seconds (Unit 2) ≤ 0.236 seconds (Unit 3)
d. Main Steam Line Flow − High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 160.5 psid (Unit 2) ≤ 117.1 psid (Unit 3)
e. Main Steam Line Tunnel Temperature − High	1,2,3	2 per trip string	D	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level − Low	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 10.24 inches
b. Drywell Pressure − High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 1.81 psig
c. Drywell Radiation − High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 77 R/hr

(continued)

Table 3.3.6.1-1 (page 2 of 3)
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
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow − High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 290.16% of rated steam flow (Unit 2) ≤ 288.23% of rated steam flow (Unit 3)
b. HPCI Steam Line Flow − Timer	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 3.2 seconds and ≤ 8.8 seconds
c. HPCI Steam Supply Line Pressure − Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 104 psig
d. HPCI Turbine Area Temperature − High	1,2,3	4[(a)]	F	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 189°F
4. Isolation Condenser System Isolation					
a. Steam Flow − High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 290.76% of rated steam flow
b. Return Flow − High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 30.2 inches water (Unit 2) ≤ 13.7 inches water (Unit 3)

(continued)

(a) All four channels must be associated with a single trip string.

Table 3.3.6.1-1 (page 3 of 3)
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
5. Reactor Water Cleanup System Isolation					
a. SLC System Initiation	1,2	1	H	SR 3.3.6.1.7	NA
b. Reactor Vessel Water Level − Low	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 10.24 inches
6. Shutdown Cooling System Isolation					
a. Recirculation Line Water Temperature − High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 346°F
b. Reactor Vessel Water Level − Low	3,4,5	2(b)	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 10.24 inches

(b) In MODES 4 and 5, provided Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

8. Turbine Stop Valve — Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve — Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 11. For this event, the reactor scram reduces the amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded.

Turbine Stop Valve — Closure signals are initiated from position switches located on each of the four TSVs. A position switch and two independent contacts are associated with each stop valve. One of the two contacts provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve — Closure channels, each consisting of one position switch (which is common to a channel in the other RPS trip system) and a switch contact. The logic for the Turbine Stop Valve — Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER $\geq$ 45% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect the OPERABILITY of this Function.

The Turbine Stop Valve — Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve — Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function even if one TSV should fail to close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is $\geq$ 45% RTP. This Function is not required when THERMAL POWER is $<$ 45% RTP since the Reactor Vessel Steam Dome Pressure — High and the Average Power Range Monitor Fixed Neutron Flux — High Functions are adequate to maintain the necessary safety margins.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Turbine Control Valve Fast Closure, Trip Oil Pressure — Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure — Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 12. For this event, the reactor scram reduces the amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure — Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER $\geq$ 45% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect the OPERABILITY of this Function. C

The Turbine Control Valve Fast Closure, Trip Oil Pressure — Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure — Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is $\geq$ 45% RTP. This Function is not required when THERMAL POWER is < 45% RTP, since the Reactor Vessel Steam Dome Pressure — High and the Average Power Range Monitor Fixed Neutron Flux — High Functions are adequate to maintain the necessary safety margins.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	12. Manual Scram (continued) Two channels of Manual Scram with one channel in each manual trip system are available and required to be OPERABLE in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.
ACTIONS	Note 1 has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPS instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel. Note 2 has been provided to modify the ACTIONS for the RPS instrumentation functions of APRM Flow Biased Neutron Flux-High (Function 2.b) and APRM Fixed Neutron Flux—High (Function 2.c) when they are inoperable due to failure of SR 3.3.1.1.2 and gain adjustments are necessary. Note 2 allows entry into associated Conditions and Required Actions to be delayed for up to 2 hours if the APRM is indicating a lower power value than the calculated power (i.e., the gain adjustment factor (GAF) is high (non-conservative)), and for up to 12 hours if the APRM is indicating a higher power value than the calculated power (i.e., the GAF is low (conservative)). The GAF for any channel is defined as the power value determined by the heat balance divided by the APRM reading for that channel. Upon completion of the gain adjustment, or expiration of the allowed time, the channel must be returned to OPERABLE status or the applicable Condition entered and the Required Actions taken. This Note is based on the time required to perform gain adjustments on multiple channels and additional time is allowed when the GAF is out of limits but conservative.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.4 and SR 3.3.1.1.8

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

As noted, SR 3.3.1.1.4 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM and APRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 24 hours after entering MODE 2 from MODE 1. Twenty four hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days for SR 3.3.1.1.4 provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 13). The Frequency of 31 days for SR 3.3.1.1.8 is acceptable based on engineering judgment, operating experience, and the reliability of this instrumentation.

SR 3.3.1.1.5

A functional test of each automatic scram contactor is performed to ensure that each automatic RPS logic channel will perform the intended function. There are four RPS channel test switches, one associated with each of the four automatic trip channels (A1, A2, B1, and B2). These test switches allow the operator to test the OPERABILITY of the individual trip logic channel automatic scram contactors as

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.5 (continued)

an alternative to using an automatic scram function trip. This is accomplished by placing the RPS channel test switch in the test position, which will input a trip signal into the associated RPS logic channel. The RPS channel test switches are not specifically credited in the accident analysis. The Manual Scram Functions are not configured the same as the generic model used in Reference 13. However, Reference 13 concluded that the Surveillance Frequency extensions for RPS Functions were not affected by the difference in configuration since each automatic RPS logic channel has a test switch which is functionally the same as the manual scram switches in the generic model. As such, a functional test of each RPS automatic scram contactor using either its associated test switch or by test of any of the associated automatic RPS Functions is required to be performed once every 7 days. The Frequency of 7 days is based on the reliability analysis of Reference 13.

SR 3.3.1.1.6 and SR 3.3.1.1.7

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required prior to fully withdrawing SRMs since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. The IRM/APRM and SRM/IRM overlaps are acceptable if a ½ decade overlap exists.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.6 and SR 3.3.1.1.7 (continued)

As noted, SR 3.3.1.1.7 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.9

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 2000 effective full power hours (EFPH) Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.10, SR 3.3.1.1.13, 3.3.1.1.15, and SR 3.3.1.1.17

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

Note 1 to SR 3.3.1.1.15 and SR 3.3.1.1.17 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. For the APRMs, changes in neutron detector sensitivity are

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.10, SR 3.3.1.1.13, SR 3.3.1.1.15, and SR 3.3.1.1.17 (continued)

compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 2000 EFPH LPRM calibration against the TIPs (SR 3.3.1.1.9). A second Note is provided that requires the APRM and IRM SRs to be performed within 24 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twenty four hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR. Note 3 to SR 3.3.1.1.15 states that for Function 2.b, this SR is not required for the flow portion of these channels. This allowance is consistent with the plant specific setpoint methodology. This portion of the Function 2.b channels must be calibrated in accordance with SR 3.3.1.1.17.

The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 31 day calibration interval in determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of a 92 day calibration interval in determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.15 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.17 is based upon the assumption of an 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.11 and SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 and SR 3.3.1.1.16 (continued)

Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.1.1.11 is based on the reliability analysis of Reference 13. The 24 month Frequency of SR 3.3.1.1.16 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.12

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 13.

SR 3.3.1.1.14

This SR ensures that scrams initiated from the Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 45% RTP. This involves calibration of the bypass channels. Adequate margins for

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.14 (continued)

the instrument setpoint methodologies are incorporated into the Allowable Value and the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during an in-service calibration at THERMAL POWER ≥ 45% RTP, if performing the calibration using actual turbine first stage pressure, to ensure that the calibration remains valid.

If any bypass channels setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 45% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 92 days is based on engineering judgment and reliability of the components.

SR 3.3.1.1.18

The LOGIC SYSTEM FUNCTIONAL TEST (LSFT) demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3, "Control Rod Operability"), and SDV vent and drain valves (LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves"), overlaps this Surveillance to provide complete testing of the assumed safety function.

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

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.19

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. This test may be performed in one measurement or in overlapping segments, with verification that all components are tested. The RPS RESPONSE TIME acceptance criteria are included in Reference 14.

As noted (Note 1), neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.

RPS RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Note 2 requires STAGGERED TEST BASIS Frequency to be determined based on 4 channels per trip system, in lieu of the 8 channels specified in Table 3.3.1.1-1 for the MSIV Closure Function. This Frequency is based on the logic interrelationships of the various channels required to produce an RPS scram signal. The 24 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. UFSAR, Section 7.2.
2. UFSAR, Section 5.2.2.2.3.
3. UFSAR, Section 6.2.1.3.2.
4. UFSAR, Chapter 15.
5. UFSAR, Section 15.4.1.
6. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
7. UFSAR, Section 15.4.10.
8. UFSAR, Section 15.6.5.

(continued)

BASES

REFERENCES (continued)

9. UFSAR, Section 15.2.5.

10. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.

11. UFSAR, Section 15.2.3.

12. UFSAR, Section 15.2.2.

13. NEDC-30851-P-A , "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

14. Technical Requirements Manual.

BASES

APPLICABLE SAFETY ANALYSES (continued)	System (RPS) Instrumentation"; IRM Neutron Flux—High and Average Power Range Monitor (APRM) Neutron Flux—High, Setdown Functions; and LCO 3.3.2.1, "Control Rod Block Instrumentation." The SRMs have no safety function and are not assumed to function during any UFSAR design basis accident or transient analysis. However, the SRMs provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in Technical Specifications.
LCO	During startup in MODE 2, three of the four SRM channels are required to be OPERABLE to monitor the reactor flux level prior to and during control rod withdrawal, subcritical multiplication and reactor criticality, and neutron flux level and reactor period until the flux level is sufficient to maintain the IRM on Range 3 or above. All but one of the channels are required in order to provide a representation of the overall core response during those periods when reactivity changes are occurring throughout the core. In MODES 3 and 4, with the reactor shut down, two SRM channels provide redundant monitoring of flux levels in the core. In MODE 5, during a spiral offload or reload, an SRM outside the fueled region will no longer be required to be OPERABLE, since it is not capable of monitoring neutron flux in the fueled region of the core. Thus, CORE ALTERATIONS are allowed in a quadrant with no OPERABLE SRM in an adjacent quadrant provided the Table 3.3.1.2-1, footnote (b), requirement that the bundles being spiral reloaded or spiral offloaded are all in a single fueled region containing at least one OPERABLE SRM is met. Spiral reloading and offloading encompass reloading or offloading a cell on the edge of a continuous fueled region (the cell can be reloaded or offloaded in any sequence). In nonspiral routine operations, two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity changes occurring in the reactor core. Because of the local nature of reactivity changes during refueling, adequate

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2.5 and SR 3.3.1.2.6

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated channel will function properly. 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. SR 3.3.1.2.5 is required in MODE 5, and the 7 day Frequency ensures that the channels are OPERABLE while core reactivity changes could be in progress. This Frequency is reasonable, based on operating experience and on other Surveillances (such as a CHANNEL CHECK), that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

C

SR 3.3.1.2.6 is required to be met in MODE 2 with IRMs on Range 2 or below, and in MODES 3 and 4. Since core reactivity changes do not normally take place in MODES 3 and 4 and core reactivity changes are due only to control rod movement in MODE 2, the Frequency is extended from 7 days to 31 days. The 31 day Frequency is based on operating experience and on other Surveillances (such as CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

Verification of the signal to noise ratio also ensures that the detectors are inserted to an acceptable operating level. In a fully withdrawn condition, the detectors are sufficiently removed from the fueled region of the core to essentially eliminate neutrons from reaching the detector. Any count rate obtained while the detectors are fully withdrawn is assumed to be "noise" only.

With few fuel assemblies loaded, the SRMs will not have a high enough count rate to determine the signal to noise ratio. Therefore, allowances are made for loading sufficient "source" material, in the form of irradiated fuel assemblies, to establish the conditions necessary to determine the signal to noise ratio. To accomplish this, SR 3.3.1.2.5 is modified by a Note that states that the

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.5 and SR 3.3.1.2.6 (continued)

determination of signal to noise ratio is not required to be met on an SRM that has less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With four or less fuel assemblies loaded around each SRM and no other fuel assemblies in the associated quadrant, even with a control rod withdrawn the configuration will not be critical.

The Note to SR 3.3.1.2.6 allows the Surveillance to be delayed until entry into the specified condition of the Applicability (THERMAL POWER decreased to IRM Range 2 or below). The SR must be performed within 12 hours after IRMs are on Range 2 or below. The allowance to enter the Applicability with the 31 day Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

SR 3.3.1.2.7

Performance of a CHANNEL CALIBRATION at a Frequency of 24 months verifies the performance of the SRM detectors and associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. The neutron detectors are excluded from the CHANNEL CALIBRATION (Note 1) because they cannot readily be adjusted. The detectors are fission chambers that are designed to have a relatively constant sensitivity over the range and with an accuracy specified for a fixed useful life.

Note 2 to SR 3.3.1.2.6 allows the Surveillance to be delayed until entry into the specified condition of the

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.7 (continued)

Applicability. The SR must be performed in MODE 2 within 12 hours of entering MODE 2 with IRMs on Range 2 or below. The allowance to enter the Applicability with the 24 month Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

REFERENCES

None.

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.

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. 12).

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

the prescribed sequence and verifying a selection error 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 at $\leq$ 10% RTP in MODE 2, and SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is $\leq$ 10% RTP in MODE 1. The Note to SR 3.3.2.1.2 allows entry into MODE 2 on a startup and entry into MODE 2 concurrent with power reduction to $\leq$ 10% RTP during a shutdown to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The Note to SR 3.3.2.1.3 allows a THERMAL POWER reduction to $\leq$ 10% RTP in MODE 1 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. Operating experience has shown that these components usually pass the Surveillance when performed at the 92 day Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.2.1.4

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

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION 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 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.

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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 24 month 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 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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.7 (continued)

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 Reactor Mode Switch — Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

C

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 24 month 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 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.2.1.8

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.9

LCO 3.1.3 and LCO 3.1.6 may require individual control rods to be bypassed (taken out of service) in the RWM to allow insertion of an inoperable control rod or correction of a control rod pattern not in compliance with the analyzed rod position sequence. With the control rods bypassed (taken out of service) in the RWM, the RWM will provide insert and withdraw blocks for bypassed control rods that are fully inserted and a withdraw block for bypassed control rods that are not fully inserted. To ensure the proper bypassing and movement of these affected control rods, 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) must verify the bypassing and position of these control rods. Compliance with this SR allows the RWM to be OPERABLE with these control rods bypassed.

REFERENCES

1. UFSAR, Section 7.6.1.5.3.
2. UFSAR, Section 7.7.2.
3. UFSAR, Section 15.4.2.3.
4. UFSAR, Section 15.4.10.
5. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1 Exxon Nuclear Methodology for Boiling Water Reactor-Neutronics Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).
6. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (as specified in Technical Specification 5.6.5).
7. Letter to T.A. Pickens (BWROG) from G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.
8. NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).

(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. 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.

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

BASES

ACTIONS

B.1 (continued)

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of Feedwater System and Main Turbine High Water Level Trip Instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1 and C.2

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 25% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below 25% RTP results in sufficient margin to the required limits, and the Feedwater System and Main Turbine High Water Level Trip Instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. Alternatively, if a channel is inoperable solely due to an inoperable feedwater pump breaker, the affected feedwater pump breaker may be removed from service since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 25% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains Feedwater System and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumptions that 6 hours is the average C

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BASES

SURVEILLANCE REQUIREMENTS (continued)

time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the feedwater pumps and main turbine will trip when necessary.

SR 3.3.2.2.1

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

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

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

SR 3.3.2.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.2.2 (continued)

Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on reliability analysis (Ref. 2).

SR 3.3.2.2.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in SR 3.3.2.2.4. If the trip setting is discovered to be less conservative than the setting accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on engineering judgement and the reliability of these components.

SR 3.3.2.2.4

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

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

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the feedwater pump breakers and main turbine stop valves is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a main turbine stop valve or feedwater pump breaker is incapable of operating, the associated instrumentation would also be inoperable. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 15.1.2.

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

BASES

LCO

4. Drywell Pressure (continued)

from separate transmitters and are continuously displayed on independent indicators in the control room. These recorders and indicators are the primary indications used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel. The drywell pressure channels also satisfy the Reference 2 monitoring requirement for suppression chamber (torus) pressure (a Type A and Category 1 variable) since the suppression chamber-to-drywell vacuum breakers ensure the suppression chamber pressure is maintained within 0.5 psig of the drywell pressure.

5. Drywell Radiation

Drywell radiation is a Category 1 variable provided to monitor the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. Two redundant radiation sensors are located in capped drywell penetrations and have a range from 10^0 R/hr to 10^8 R/hr. These radiation monitors display on recorders located in the control room. Two radiation monitors/recorders are required to be OPERABLE (one per division). Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

6. Penetration Flow Path Primary Containment Isolation Valve (PCIV) Position

PCIV (excluding check valves) position is a Category 1 variable provided for verification of containment integrity. In the case of PCIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active PCIV in a containment penetration flow path requiring post-accident valve position indication, i.e., two total channels of PCIV position indication for a penetration flow path with two active valves requiring post-accident valve position indication. For containment penetrations with only one

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.2

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in SR 3.3.4.1.4. If the trip setting is discovered to be less conservative than the setting accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the ATWS analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on engineering judgement and the reliability of these components.

SR 3.3.4.1.3

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

The Frequency of 92 days is based on the reliability analysis of Reference 3.

SR 3.3.4.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor, including the time delay relays associated with the Reactor Vessel Water Level — Low Low Function. This test verifies the channel responds to the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1.4 (continued)

measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

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

SR 3.3.4.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the pump breakers is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel(s) would be inoperable.

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

REFERENCES

1. UFSAR, Section 7.8.

2. UFSAR, Section 15.8

3. GENE-770-06-1-A, "Bases for Changes To Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications," December 1992.

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

The LPCI System initiation logic also contains LPCI Loop Select Logic whose purpose is to identify and direct LPCI flow to the unbroken recirculation loop if a Design Basis Accident (DBA) occurs. The LPCI Loop Select Logic is initiated upon the receipt of either a LPCI Reactor Vessel Water Level — Low Low signal or a LPCI Drywell Pressure — High signal, as discussed previously. When initiated, the LPCI Loop Select Logic first determines recirculation pump operation by sensing the differential pressure (dp) between the suction and discharge of each pump. There are four dp switches monitoring each recirculation loop which are, in turn, connected to relays whose contacts are connected to two trip systems. The dp switches will trip when the dp across the pump is approximately 2 psid. The contacts are arranged in a one-out-of-two taken twice logic for each recirculation pump. If the logic senses that either pump is not running, i.e., single loop operation, then a trip signal is sent to both recirculation pumps to eliminate the possibility of pipe breaks being masked by the operating recirculation pump pressure. However, the pump trip signal is delayed approximately 0.5 seconds (one time delay relay per trip system) to ensure that at least one pump is off since the break detection sensitivity is greater with both pumps running. If a pump trip signal is generated, reactor steam dome pressure must drop to a specified value before the logic will continue. This adjusts the selection time to optimize sensitivity and still ensure that LPCI injection is not unnecessarily delayed. The reactor steam dome pressure is sensed by four pressure switches which in turn are connected to relays whose contacts are connected to two trip systems. The contacts are arranged in a one-out-of-two taken twice logic. After the satisfaction of this pressure requirement or if both recirculation pumps indicate they are running, a 2 second time delay is provided to allow momentum effects to establish the maximum differential pressure for loop selection. Selection of the unbroken recirculation loop is then initiated. This is done by comparing the absolute pressure of the two recirculation riser loops. The broken loop is indicated by a lower pressure than the unbroken loop. The loop with the higher pressure is then used for LPCI injection. If, after a small time delay of approximately 0.5 seconds (one time delay relay per trip system), the pressure in loop A is not indicating

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.2

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

The Frequency of 92 days is based on the reliability analyses of Reference 5.

SR 3.3.5.1.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.5.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analyses. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than the setting accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 5.

SR 3.3.5.1.4 and SR 3.3.5.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.4 and SR 3.3.5.1.5 (continued)

adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.5.1.4 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.5.1.5 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety function.

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

REFERENCES

1. UFSAR, Section 5.2.
2. UFSAR, Section 6.3.
3. UFSAR, Chapter 15.
4. EMF-97-025(P), Revision 1, "LOCA Break Spectrum Analysis for Dresden Units 2 and 3," May 30, 1997.
5. NEDC-30936-P-A, "BWR Owners' Group Technical Specification Improvement Analyses for ECCS Actuation Instrumentation, Part 1 and Part 2," December 1988.

BASES

ACTIONS

A.1 and A.2 (continued)

Because of the redundancy of sensors available to provide initiation signals and the fact that the IC System is not assumed in any accident or transient analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition B must be entered and its Required Action taken.

B.1

With any Required Action and associated Completion Time of Condition A not met, the IC System may be incapable of performing the intended function, and the IC System must be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the Reactor Vessel Pressure—High Function maintains initiation 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.

SR 3.3.5.2.1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.2.1 (continued)

change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 31 days is based on plant operating experience with regard to channel OPERABILITY and drift that demonstrates that failure of more than one channel in any 31 day interval is rare.

SR 3.3.5.2.2 and SR 3.3.5.2.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. A Note to SR 3.3.5.2.2 states that this SR is not required for the time delay portion of these channels. This allowance is consistent with the plant specific setpoint methodology. This portion of the channels must be calibrated in accordance with SR 3.3.5.2.3.

The Frequency of SR 3.3.5.2.2 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.5.2.3 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.3 overlaps this Surveillance to provide complete testing of the safety function.

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

REFERENCES

1. GENE-770-06-2-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

BASES

BACKGROUND (continued)

1. Main Steam Line Isolation

The Reactor Vessel Water Level — Low Low, Main Steam Line Pressure — Low, and Main Steam Line Pressure — Timer Functions receive inputs from four channels. One channel associated with each Function inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of all main steam isolation valves (MSIVs), MSL drain valves, and recirculation loop sample isolation valves. Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation. 1C

The Main Steam Line Flow — High Function uses 16 flow channels, four for each steam line. One channel from each steam line inputs to one of the four trip strings. Two trip strings make up each trip system and both trip systems must trip to cause an isolation of all MSIVs, MSL drain valves, and recirculation sample isolation valves. Each trip string has four inputs (one per MSL), any one of which will trip the trip string. The trip strings are arranged in a one-out-of-two taken twice logic. This is effectively a one-out-of-eight taken twice logic arrangement to initiate isolation.

The Main Steam Line Tunnel Temperature — High Function receives input from 16 channels, four for each of the four tunnel areas. The logic is arranged similar to the Main Steam Line Flow — High Function. One channel from each steam tunnel area inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation.

MSL Isolation Functions isolate the Group 1 valves.

2. Primary Containment Isolation

The Reactor Vessel Water Level — Low and Drywell Pressure — High Functions receive inputs from four channels. One channel associated with each Function inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the PCIVs identified in Reference 1. Any channel will trip the

(continued)

BASES

BACKGROUND

3, 4. High Pressure Coolant Injection System Isolation and Isolation Condenser System Isolation (continued)

The HPCI Steam Supply Line Pressure—Low Function receives input from four steam supply pressure channels. The outputs from the HPCI steam supply pressure channels are connected in a one-out-of-two-twice arrangement which provides input to two trip systems. Either trip system isolates both valves in the HPCI steam supply penetration.

The HPCI Turbine Area Temperature—High Function receives input from 16 temperature switches. Four channels, each with an associated temperature switch, provide inputs to a one-out-of-two-twice logic arrangement in each of two AC and two DC trip strings. Each of the trip strings provides input into both an AC and DC trip system and only one trip string must trip to trip the associated trip system. However for OPERABILITY, only one DC trip string is required to provide input into the DC trip system and only one AC trip string is required to provide input into the AC trip system. Either trip system isolates both valves in the HPCI steam supply penetration.

C

HPCI and Isolation Condenser Functions isolate the Group 4 and 5 valves, as appropriate.

5. Reactor Water Cleanup System Isolation

The Reactor Vessel Water Level—Low Isolation Function receives input from four reactor vessel water level channels. Each channel inputs into one of four trip stings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the reactor water cleanup (RWCU) valves. Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation. The SLC System Initiation Function receives input from the SLC initiation switch. The switch provides trip signal inputs to both trip systems in any position other than "OFF". The other switch positions are SYS 1, SYS 2, SYS 1+2 and SYS 2+1. For the purpose of this Specification, the SLC

C

(continued)

BASES

BACKGROUND

5. Reactor Water Cleanup System Isolation (continued)

initiation switch is considered to provide 1 channel input into each trip system. Each of the two trip systems is connected to one of the two RWCU valves.

RWCU Functions isolate the Group 3 valves.

6. Shutdown Cooling (SDC) System Isolation

The Reactor Vessel Water Level — Low Function receives input from four reactor vessel water level channels. Each channel inputs into one of four trip stings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the SDC suction isolation valves. Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation. The Recirculation Line Water Temperature — High Function receives input from two channels, both of which provide input to both trip systems. Any channel will trip both trip systems. This is a one-out-of-two logic for each trip system. Each of the two trip systems is connected to one of the two valves on the SDC suction penetration. Only one of the trip systems isolates the SDC return penetration. C

Shutdown Cooling System Isolation Functions isolate some Group 3 valves (SDC isolation valves).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the primary containment isolation instrumentation are implicitly assumed in the safety analyses of References 2 and 3 to initiate closure of valves to limit offsite doses. Refer to LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Applicable Safety Analyses Bases for more detail of the safety analyses.

Primary containment isolation instrumentation satisfies Criterion 3 of 10 CFR 50.35(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>1.b. Main Steam Line Pressure — Low</u> (continued)

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

The Main Steam Line Pressure — Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 5).

This Function isolates the Group 1 valves.

<u>1.c. Main Steam Line Pressure — Timer</u>

The Main Steam Line Pressure — Timer is provided to prevent false isolations on low MSL pressure as a result of pressure transients, however, the timer must function in a limited time period to support the OPERABILITY of the Main Stem Line Pressure — Low Function by enabling the associated channels after a certain time delay. The Main Steam Line Pressure — Timer is directly assumed in the analysis of the pressure regulator failure (Ref. 5). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded.

The MSL low pressure timer signals are initiated when the associated MSL low pressure switch actuates. Four channels of Main Steam Line Pressure — Timer Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to be long enough to prevent false isolations due to pressure transient but short enough to prevent excessive RPV depressurizations.

This Function isolates the Group 1 valves.

<u>1.d. Main Steam Line Flow — High</u>

Main Steam Line Flow — High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.c. HPCI Steam Supply Line Pressure – Low (continued)

The Allowable Values are selected to be high enough to prevent damage to the system turbine.

These Functions isolate the Group 4 valves.

3.d. HPCI Turbine Area Temperature – High

HPCI turbine area temperatures are provided to detect a leak from the HPCI system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. These Functions are not assumed in any UFSAR transient or accident analysis, since bounding analyses are performed for large breaks such as recirculation or MSL breaks.

HPCI Turbine Area Temperature – High signals are initiated from temperature switches that are appropriately located to detect a leak from the system piping that is being monitored. Four instruments monitor each area. Sixteen instruments monitor the HPCI Turbine Area. Sixteen channels for HPCI Turbine Area Temperature – High are available, however only eight channels are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. As noted (footnote (a) to Table 3.3.6.1-1) each trip system associated with this Function requires all four channels to be associated with a single trip string (four channels within the same AC trip string for the AC trip system and four channels within the same DC trip string for the DC trip system).

The Allowable Value is set well above the expected ambient condition but low enough to detect steam line leakage.

These Functions isolate the Group 4 valves.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.1

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

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

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

SR 3.3.6.1.2 and SR 3.3.6.1.5

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.2 and SR 3.3.6.1.5 (continued)

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analyses described in References 8 and 9. The 24 month Frequency of SR 3.3.6.1.5 is based on engineering judgement and the reliability of the components.

SR 3.3.6.1.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.6.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than that accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of References 9 and 10.

SR 3.3.6.1.4 and SR 3.3.6.1.6

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

The Frequency of SR 3.3.6.1.4 is based on the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.6.1.6 is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.7

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. Technical Requirements Manual.
2. UFSAR, Section 6.2.
3. UFSAR, Chapter 15.
4. UFSAR, Section 15.6.5.
5. UFSAR, Section 15.1.3.
6. UFSAR, Section 15.6.4.
7. UFSAR, Section 9.3.5.
8. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
9. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.2.1

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

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

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

SR 3.3.6.2.2

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2.2 (continued)

The Frequency of 92 days is based on the reliability analysis of References 4 and 5.

SR 3.3.6.2.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.6.2-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of References 4 and 5.

SR 3.3.6.2.4 and SR 3.3.6.2.5

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

The Frequencies of SR 3.3.6.2.4 and SR 3.3.6.2.5 are based on the assumption of a 92 day and a 24 month calibration interval, respectively, in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.3.6.2.6 The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on SCIVs and the SGT System in LCO 3.6.4.2 and LCO 3.6.4.3, respectively, overlaps this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.
REFERENCES	1. UFSAR, Section 6.2.3. 2. UFSAR, Section 15.6.5. 3. UFSAR, Section 15.7.3. 4. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990. 5. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

BASES

BACKGROUND

The CREV System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. The CREV System is capable of fulfilling the stated safety function. The CREV System instrumentation provides control room alarms so that manual action can be taken to start the CREV System and pressurize control room emergency zone to minimize the consequences of radioactive material in the control room environment.

In the event of a Reactor Building Ventilation System—High High Radiation alarm signal, operator action is required to switch the CREV System to the isolation/pressurization mode of operation and close required dampers to maintain the control room emergency zone slightly pressurized with respect to the adjacent zones. A description of the CREV System is provided in the Bases for LCO 3.7.4, "Control Room Emergency Ventilation (CREV) System."

The CREV System instrumentation has two trip systems, either of which provide sufficient information to ensure the CREV System is initiated and the dampers are closed when necessary. Each trip system receives input from one radiation monitor channel. Two detectors (one detector for each radiation monitor channel) are located in the reactor building ventilation exhaust duct. The output of each channel is provided to one trip system (i.e., one radiation monitor channel per trip system). The output from each channel is arranged in a one-out-of-one trip (alarm) system. A trip of any trip system will initiate a Reactor Building Ventilation System—High High Radiation Alarm in the control room. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a signal to the alarm logic.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ability of the CREV System to maintain the habitability of the control room emergency zone is explicitly assumed for certain accidents as discussed in the UFSAR safety analyses (Refs. 1, and 2). CREV System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A.

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

LCO

High reactor building ventilation exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When high reactor building ventilation exhaust radiation is alarmed in the control room, the CREV System is manually initiated in the isolation/pressurization mode and required dampers are closed since this condition could result in radiation exposure to control room personnel.

The Reactor Building Ventilation System—High High Radiation Alarm Function signals are initiated from radiation detectors that are located in the ventilation exhaust ducting coming from the reactor building and refueling zones. The signals from each detector are input to individual monitors whose trip outputs are assigned to a control room alarm. Two channels of Reactor Building Ventilation System—High High Radiation Alarm Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the alarm function. The Allowable Value was selected to promptly detect gross failure of the fuel cladding and to ensure protection of control room personnel. Each channel must have its setpoint set within the specified Allowable Value in SR 3.3.7.1.3. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. 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 successive CHANNEL

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.2

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

The Frequency of 92 days is based on the reliability analyses of References 3 and 4.

SR 3.3.7.1.3

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

The Frequency is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Section 6.4.

2. UFSAR, Section 15.6.5.

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

4. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.1.1 and SR 3.3.8.1.3

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

The Frequencies of 18 months and 24 months are based on operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 18 month or 24 month interval, as applicable, is a rare event.

SR 3.3.8.1.2 and SR 3.3.8.1.4

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

The Frequency is based upon the assumption of an 18 month or 24 month calibration interval, as applicable, in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	<u>SR 3.3.8.1.5</u> The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The system functional testing performed in LCO 3.8.1 and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety functions. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.
REFERENCES	1. UFSAR, Section 8.3.1.7. 2. UFSAR, Section 5.2. 3. UFSAR, Section 6.3. 4. UFSAR, Chapter 15.

BASES

BACKGROUND (continued)

circuit breakers has an associated independent set of Class 1E overvoltage, undervoltage, and underfrequency sensing logic. Together, a circuit breaker and its sensing logic constitute an electric power monitoring assembly. If the output of the inservice MG set or alternate power supply exceeds predetermined limits of overvoltage, undervoltage, or underfrequency, a trip coil (undervoltage release coil) within the circuit breaker driven by this logic circuitry opens the circuit breaker, which removes the associated power supply from service.

APPLICABLE SAFETY ANALYSES

The RPS Electric Power Monitoring is necessary to meet the assumptions of the safety analyses by ensuring that the RPS equipment powered from the RPS buses can perform its intended function. RPS Electric Power Monitoring provides protection to the RPS components, by acting to disconnect the RPS bus from the power supply under specified conditions that could damage the RPS equipment.

RPS Electric Power Monitoring satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of each RPS electric power monitoring assembly is dependent on the OPERABILITY of the overvoltage, undervoltage, and underfrequency logic, as well as the OPERABILITY of the associated circuit breaker. Two electric power monitoring assemblies are required to be OPERABLE for each inservice power supply. This provides redundant protection against any abnormal voltage or frequency conditions to ensure that no single RPS electric power monitoring assembly failure can preclude the function of RPS bus powered components. Each of the inservice electric power monitoring assembly trip logic setpoints is required to be within the specified Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each RPS electric power monitoring assembly trip logic (refer to SR 3.3.8.2.2). Nominal trip setpoints are specified in the setpoint

(continued)

BASES

ACTIONS (continued)

D.1

If any Required Action and associated Completion Time of Condition A or B are not met in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, the operator must immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Required Action D.1 results in the least reactive condition for the reactor core and ensures that the safety function of the RPS (e.g., scram of control rods) is not required.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.2.1

A CHANNEL FUNCTIONAL TEST is performed on each overvoltage, undervoltage, and underfrequency channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. C

As noted in the Surveillance, the CHANNEL FUNCTIONAL TEST is only required to be performed while the plant is in a condition in which the loss of the RPS bus will not jeopardize steady state power operation (the design of the system is such that the power source must be removed from service to conduct the Surveillance). The 24 hours is intended to indicate an outage of sufficient duration to allow for scheduling and proper performance of the Surveillance.

The 184 day Frequency and the Note in the Surveillance are based on guidance provided in Generic Letter 91-09 (Ref. 2).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.2.2

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

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

SR 3.3.8.2.3

Performance of a system functional test demonstrates that, with a required system actuation (simulated or actual) signal, the logic of the system will automatically trip open the associated power monitoring assembly. The system functional test shall include actuation of the protective relays, tripping logic, and output circuit breakers. Only one signal per power monitoring assembly is required to be tested. This Surveillance overlaps with the CHANNEL CALIBRATION to provide complete testing of the safety function. The system functional test of the Class 1E circuit breakers is included as part of this test to provide complete testing of the safety function. If the breakers are incapable of operating, the associated electric power monitoring assembly would be inoperable.

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

REFERENCES

1. UFSAR, Section 7.2.3.

2. NRC Generic Letter 91-09, "Modification of Surveillance Interval for the Electrical Protective Assemblies in Power Supplies for the Reactor Protection System."

REACTOR PROTECTION SYSTEM (3.3.1.1-1) RPS 3/4.1.A

TABLE ~~4.1.A-1~~ (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

Note 1 SR 3.3.1.1.15 SR 3.3.1.1.17 (a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

SR 3.3.1.1.6 (b) The IRM and SRM channels shall be determined to overlap ~~for at least (½) decades~~ during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap ~~for at least (½) decades~~ during each controlled shutdown, if not performed within the previous 7 days. LA.3

SR 3.3.1.1.7 A.12

(c) ~~Within 24 hours prior to startup, if not performed within the previous 7 days.~~ The weekly SR 3.3.1.1.4 CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) SR 3.3.1.1.2 This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. ACTIONS Note 2 This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. ~~Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.~~ L.11

SR 3.3.1.1.2 Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. ~~This calibration is not required when THERMAL POWER is < 25% of RATED THERMAL POWER.~~ The provisions of Specification 4.0.D are not applicable. SR 3.3.1.1.2 Note not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP L.7 C

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to SR 3.3.1.1.3 a calibrated flow signal.

SR 3.3.1.1.9 (f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).

~~(g) Deleted.~~ A.3

SR 3.3.1.1.12 SR 3.3.1.1.17 (h) Trip units are calibrated at least once per ~~31~~ 92 days and transmitters are calibrated at the frequency identified in the table.

~~(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.~~ from a core cell containing one or more fuel assemblies A.6 L.4

Table 3.3.1.1-1 footnote (a) (j) With any control rod withdrawn. ~~Not applicable to control rods removed per Specification 3.10.I or 3.10.J.~~ A.9

~~(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.~~ L.4

A.1

3.10 - LIMITING CONDITIONS FOR OPERATION

B. Instrumentation

LCO 3.3.1.2 and Table 3.3.1.2-1 At least 2 source range monitor[a] (SRM) CHANNEL(s) shall be OPERABLE ~~and inserted to the normal operating level with:~~ LA.2

~~1. Continuous visual indication in the control room, and~~

SR 3.3.1.2.2.b and c 2. One of the required SRM detectors located in the quadrant where CORE ALTERATION(s) are being performed and the other required SRM detector located in an adjacent quadrant.

M.5 add proposed SR 3.3.1.2.5 Note

APPLICABILITY:

add proposed Note b to Table 3.3.1.2-1 L.7

Table 3.3.1.2-1 OPERATIONAL MODE 5~~, unless the following conditions are met:~~

~~1. No more than two fuel assemblies are present in each core quadrant associated with an SRM.~~ M.2

4.10 - SURVEILLANCE REQUIREMENTS

B. Instrumentation

Each of the required SRM channels shall be demonstrated OPERABLE by:

1. At least once per 12 hours:

   SR 3.3.1.2.1 a. Performance of a CHANNEL CHECK.

   ~~b. Verifying the detectors are inserted to the normal operating level, and~~ LA.2

   M.3 add proposed SR 3.3.1.2.2.a and SR 3.3.1.2.2 Note 2

   SR 3.3.1.2.2 Note 1 c. During CORE ALTERATION(s), verifying that the detector of an OPERABLE SRM CHANNEL is located in the core quadrant where SR 3.3.1.2.2.b and c CORE ALTERATION(s) are being performed and another is located in an adjacent quadrant.

SR 3.3.1.2.5 2. Performance of a CHANNEL FUNCTIONAL TEST~~:~~ M.5 and determination of signal to noise ratio

   ~~a. Within 24 hours prior to the start of CORE ALTERATION(s), and~~ L.3

   b. At least once per 7 days. L.4 add proposed SR 3.3.1.2.4 Note

SR 3.3.1.2.4 3. Verifying that the channel count rate is at least 3 cps~~:~~ or ≥ 0.7 cps with a signal to noise ratio ≥ 20:1 L.8

   ~~a. Prior to control rod withdrawal,~~ L.3

   b. Prior to and at least once per 12 hours during CORE ALTERATION(s),

   c. At least once per 24 hours.

add proposed SR 3.3.1.2.7 M.4

Table 3.3.1.2-1 Note c a The use of special movable detectors during CORE ALTERATION(s) in place of the normal SRM neutron detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.2 CTS 3.10.B Applicability provides exceptions to the MODE 5 requirements to maintain at least 2 source range monitor (SRM) channels OPERABLE. CTS 3.10.B Applicability does not require SRMs to be OPERABLE when no more than two fuel assemblies are present in each core quadrant with an SRM when those fuel assemblies are positioned adjacent to that quadrant's SRM. CTS 3.10.B also provides specific criteria to be met if movable detectors are being used (see Discussion of Change LA.3). Proposed ITS 3.3.1.2 requires at least two SRM channels to be OPERABLE when in MODE 5 (unless performing a spiral offload or reload), but provides specific allowances in verifying OPERABILITY for conditions when the removal of fuel assemblies would not maintain the required count rate in proposed SR 3.3.1.2.4 and provides specific verification requirements for the positioning of the required OPERABLE SRM detectors in SR 3.3.1.2.2. These Surveillance Requirements encompass the allowances specified in the CTS 3.10.B Applicability. This change represents an additional restriction on plant operation necessary to ensure the SRMs are capable of monitoring reactivity changes in the core during refueling. 1C

M.3 CTS 4.10.B.1.a requires verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and one is located in the adjacent quadrant. ITS SR 3.3.1.2.2 requires verifying that an OPERABLE SRM detector is located in the fueled region; the core quadrant where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region; and in a core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region. As a result of providing the additional criteria on where the OPERABLE SRMs must be relocated, Note 2 to ITS SR 3.3.1.2.2 is also added to clarify that more than one of the three requirements of ITS SR 3.3.1.2.2 can be satisfied by the same SRM since only two SRMs are required to be OPERABLE. Providing additional criteria on where the SRMs must be located to satisfy the Surveillance represents an additional restriction on plant operation necessary to provide adequate coverage of potential reactivity changes in the core and to achieve consistency with NUREG-1433, Revision 1.

M.4 A new Surveillance Requirement has been added, proposed SR 3.3.1.2.7, requiring the SRMs to be calibrated every 24 months if in MODE 5. This SR verifies the performance of the SRM detectors and associated circuitry. This is an additional restriction on plant operation necessary to ensure the OPERABILITY of the SRMs during MODE 5.

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 calendar year (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.

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.

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.

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

LA.1 (cont'd) instrumentation since OPERABILITY requirements are adequately addressed in ITS 3.3.2.1. In addition, when a peripheral control rod is selected, RBM is automatically bypassed and cannot generate a rod block. Therefore, the Applicabilities for the RBM Functions have been modified to be ≥ 30% RTP and no peripheral control rod selected, consistent with the design and CTS Table 3.2.E-1 Note (a) (see Discussion of Change A.3 above). 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 UFSAR will be controlled by the provisions of 10 CFR 50.59.

LA.2 Details in Table 4.2.E-1 Function 1 footnote c, CTS 4.3.L.2.a and b, and CTS 4.3.L.3.a and b of the methods for performing Surveillances are proposed to be relocated to the Bases. The requirements proposed to be relocated are procedural details that are not necessary for assuring control rod block instrumentation OPERABILITY. The Surveillance Requirements of ITS 3.3.2.1 provide adequate assurance the control rod block instrumentation are maintained OPERABLE. As such, 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.

LF.1 This change revises the Current Technical Specifications (CTS) Trip Setpoints for the Improved Technical Specifications (ITS) Allowable Values. ITS Section 3.3 reflects Allowable Values consistent with the philosophy of BWR ISTS, NUREG-1433, 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") or NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996 (for Nuclear Instrumentation System Functions only). 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 A

TECHNICAL CHANGES - LESS RESTRICTIVE

LF.1 (cont'd) 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 ComEd or General Electric (GE) Instrument Setpoint Methodology. The Allowable Values have been established from each design or safety analysis limit 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 ComEd or GE 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, or the methodology described in NEDC-31336P-A (for Nuclear Instrumentation System Functions only). The EPRI guidance and GE methodology were 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 Surveillance Frequency of "S/U" and Note (b), "within 7 days prior to startup," associated with the CHANNEL FUNCTIONAL TEST of the RBM Functions in CTS Table 4.3.E-1 is deleted. The requirements of CTS 4.0.A and 4.0.D (ITS SR 3.0.1 and SR 3.0.4) require the Surveillance to be performed and current prior to entry into the applicable Operational Conditions. Additionally, once the applicable Conditions are entered, the periodic Surveillance Frequency (92) days) has been determined to provide adequate assurance of RBM OPERABILITY per the reliability analysis of NEDO-30851P-A, "Technical Specifications Improvement Analysis for BWR Control Rod Block

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) Instrumentation," dated October 1988. Also, the increased testing prior to startup increases the wear on the instruments, thereby reducing overall reliability. Therefore, an additional Surveillance other than the quarterly Surveillance (ITS SR 3.3.2.1.1) is not needed to assure the instruments will perform their associated safety function.

L.2 CTS 4.3.L.2 requires a RWM CHANNEL FUNCTIONAL TEST to be performed within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical and CTS 4.3.L.3 requires a RWM CHANNEL FUNCTIONAL TEST to be performed prior to reducing thermal power to < 20% RTP. Proposed SRs 3.3.2.1.2 and 3.3.2.1.3 are similar to CTS 4.3.L.2 and 4.3.L.3, except a test Frequency is specified (92 days). This change effectively extends the CHANNEL FUNCTIONAL TEST to 92 days, i.e., the CHANNEL FUNCTIONAL TEST is not required to be performed if a startup or shutdown occurs within 92 days of a previous startup or shutdown. The RWM is a reliable system, as shown by both a review of maintenance history and by successful completion of previous startup surveillances. As a result, the effect on safety due to the extended Surveillance will not be significant. Also, the increased testing prior to each startup and shutdown increases the wear on the instruments, thereby reducing overall reliability. Therefore, an additional Surveillance other than the quarterly Surveillance is not needed to assure the instruments will perform their associated safety function. In addition, other similar rod block functions have a 92 day CHANNEL FUNCTIONAL TEST. Notes are also being added to CTS 4.3.L.2 and 3. The Note to proposed SR 3.3.2.1.2 exempts the CHANNEL FUNCTIONAL TEST requirement of the RWM until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. The Note to proposed SR 3.3.2.1.3 exempts the CHANNEL FUNCTIONAL TEST requirement of the RWM until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. These changes are acceptable since the only way the required Surveillances can be performed prior to entry in the specified condition is by utilizing jumpers or lifted leads. Use of these devices is not recommended since minor errors in their use may significantly increase the probability of a reactor transient or event which is a precursor to a previously analyzed accident. Therefore, time is allowed to conduct the Surveillances after entering the specified condition.

L.3 CTS 3.3.M Action 1.a, which requires verification that the reactor is not operating on a LIMITING CONTROL ROD PATTERN when one RBM channel is inoperable, and Surveillance Requirement 4.3.M.2, which requires a CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the

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 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 consequences of a CRDA above 10% RTP are mitigated by factors which reduce available rod worths and

TECHNICAL CHANGES - LESS RESTRICTIVE

L.4 (cont'd) 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.3 CTS Table 3.2.F-1 ACTIONS 60a and 62a, for one channel inoperable in one or more Functions for more than the allowed outage time is revised from requiring a shutdown to requiring a Special Report (ITS 3.3.3.1 Required Action B.1) in accordance with the Administrative Control section of the Technical Specifications. Due to the passive function of these instruments and the operator's ability to respond to an accident utilizing alternate instruments and methods for monitoring, it is not appropriate to impose stringent shutdown requirements for out of service instrumentation. The change is considered acceptable since another OPERABLE channel is monitoring the Function and the probability of an event, requiring the operator to utilize this instrumentation to respond to the event, is low. This change is consistent with the BWR ISTS, NUREG-1433, Rev. 1.

L.4 The CTS Table 3.2.F-1 ACTION 60b allowable outage time for restoration of two inoperable monitors is revised from 48 hours to 7 days in proposed ITS 3.3.3.1 Required Action C.1. Due to the passive nature of these instruments and the operator's ability to respond to an accident utilizing alternate instruments and methods of monitoring, it is not appropriate to impose stringent shutdown requirements for out-of-service instrumentation. The change is considered acceptable since an alternate method of monitoring the Function is available and the probability of an event, requiring the operator to utilize this instrumentation to respond to the event, is low. This change is consistent with BWR ISTS, NUREG-1433, Rev. 1.

L.5 CTS Table 3.2.F-1 ACTION 61 is changed for one or two drywell area radiation monitors inoperable. With one monitor inoperable, ITS 3.3.3.1 Required Action A.1 provides 30 days for the restoration of the monitor prior to initiating action in accordance with Specification 5.6.6. With two monitors inoperable, ITS 3.3.3.1 Required Action C.1 provides 7 days for restoration of one monitor prior to initiating the alternate method of monitoring. With one or two monitors inoperable CTS Table 3.2.F-1 ACTION 61 requires initiation of the alternate method of monitoring within 72 hours and restoration of both channels to OPERABLE status within 7 days. The Completion Times (30 days when one monitor is inoperable or 7 days when two monitors are inoperable) for restoration of one channel or initiation of action in accordance with Specification 5.6.6 is considered acceptable based on the relatively low probability of an event requiring PAM instrumentation, the passive function of the instruments, the availability of the redundant monitor (for the condition of one monitor inoperable), and the availability of alternate means to obtain the information.

ADMINISTRATIVE

A.4 (cont'd) trip setpoints specified in CTS Table 3.2.C-1 for the ATWS-RPT instrumentation Functions or the Allowable Values specified in ITS 3.3.4.1 (see Discussion of Change LF.1 below for proposed changes to the trip setpoints/Allowable Values). Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

A.5 The Trip Setpoint for Functional Unit 1, Reactor Vessel Water Level - Low Low, in Table 3.2.C-1 is referenced to the top of active fuel. The reference value for the Allowable Value specified in ITS SR 3.3.4.1.4.a is associated with "instrument zero." This change has been made for human factors considerations. The indications in the control room can be directly associated with the value in the ITS. Any change to the Trip Setpoint is addressed in Discussion of Changes A.4 and LF.1, therefore this change is considered administrative. A

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The ATWS trip logic uses a two-out-of-two logic for each trip Function in both trip systems. The reactor recirculation pumps will trip when one trip system actuates. Therefore, when a channel associated with one Trip Function (e.g., Reactor Water Level - Low Low) is inoperable in both trip systems, the ATWS-RPT trip capability is lost for that Function. Similarly, if channels associated with both Trip Functions are inoperable in both trip systems, the ATWS-RPT trip capability is lost for both ATWS-RPT trip Functions. CTS 3.2.C Action 2 and 4 address the condition with channels inoperable in both trip systems. Under these conditions the ATWS-RPT trip capability is lost for one and two Trip Functions, respectively. In the ITS, these conditions will require entry into proposed ITS 3.3.4.1 ACTION B and ACTION C, respectively. The Completion Times (72 hours and 1 hour, respectively) are consistent with the current actions for loss of trip function capability in CTS 3.2.C Actions 5 and 6, respectively. Since the current allowances have been deleted, this change is considered more restrictive on plant operations but necessary to limit the time the plant is allowed to operate with a loss of trip capability.

M.2 If the channels are inoperable due to a trip breaker that will not open, placing the channels in the tripped condition, as required by CTS 3.2.C Action 2 will not accomplish the intended restoration of the functional capability. Therefore, a Note is added to ITS 3.3.4.1 Required Action A.2 to prevent proposed Required

TECHNICAL CHANGES - MORE RESTRICTIVE

M.2 (cont'd) Action A.2 (place channel in trip) from being used in these conditions. This new Note will ensure the functional capability of the ATWS-RPT is restored (by restoring the inoperable channel) within the allowed Completion Time when a trip breaker is inoperable. In addition, the LOGIC SYSTEM FUNCTIONAL TEST in CTS 4.2.C.2 (proposed ITS SR 3.3.4.1.5) has been revised to include breaker actuation. This added requirement will ensure the complete testing of the assumed function. These changes are more restrictive on plant operation and necessary to ensure that ATWS-RPT Functions are adequately maintained.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details in CTS 3.2.C Action 2 footnote (a), relating to placing channels in trip, are proposed to be relocated to the Bases. The ACTIONS of ITS 3.3.4.1 ensure inoperable channels are placed in trip or the unit is placed in a non-applicable MODE or condition, as appropriate. In addition, the Bases for Required Actions A.1 and A.2 indicate that the channels are not required to be placed in the trip condition, and directs entry into the appropriate Condition. As a result, these relocated details are not necessary for ensuring the appropriate actions are taken in the event of inoperable ATWS-RPT Instrumentation channels. As such, these 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. Q

LA.2 The detail in CTS Table 3.2.C-1 Note (c) related to the reference setting of the reactor vessel water level instrumentation is proposed to be relocated to the UFSAR. The reference value for the Allowable Value specified in ITS SR 3.3.4.1.4.a has been changed to the value associated with "instrument zero," as discussed in Discussion of Change A.5. This detail is not necessary to ensure the OPERABILITY of the ATWS-RPT instrumentation. The requirements of ITS 3.3.4.1 and the Surveillances are adequate to ensure the ATWS-RPT reactor vessel water level instrumentation is maintained OPERABLE. Therefore, this relocated detail is not required to be in the ITS to provide adequate protection of public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59. A

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LD.1 The Frequency for performing the LOGIC SYSTEM FUNCTIONAL TEST of CTS 4.2.C.2 (proposed SR 3.3.4.1.5) has been extended from 18 months to 24 months. This SR ensures that ATWS-RPT System will function as designed to ensure proper response during an analyzed event. 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 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 this test normally passes the Surveillance 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. Extending the SR interval for this function is acceptable because the ATWS-RPT logic is tested every 92 days by the Channel Functional Test in CTS 4.2.C.1 and Table 4.2.C-1 (proposed SR 3.3.4.1.3). This testing of the ATWS-RPT System ensures that a significant portion of the circuitry is operating properly and will detect significant failures of this circuitry. The ATWS-RPT System including the actuating logic is designed to be single failure proof and therefore, is highly reliable. 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. This historical 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 small from a change to a 24 month

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) operating cycle. 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.

LE.1 The Frequency for performing the CHANNEL CALIBRATION Surveillance of CTS 4.2.C.1 and Table 4.2.C-1 Trip Functions 1 and 2 (proposed SR 3.3.4.1.4) has been extended from 18 months to 24 months. 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.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 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. The CHANNEL CALIBRATION Surveillance is performed to ensure that a previously evaluated setpoint actuation takes place to provide the required safety function. Extending the SR Frequency is acceptable because the ATWS-RPT initiation logic is designed to be single failure proof, and therefore, is highly reliable. Furthermore, the impacted ATWS-RPT instrumentation has been evaluated based on make, manufacturer and model number to determine that the instrumentation's actual drift falls within the design allowance in the associated setpoint calculation. The following paragraphs, listed by CTS Functional Unit, identify by make, manufacturer and model number the drift evaluations performed:

Functional Unit 1, Reactor Vessel Water Level - Low Low

This function is performed by Rosemount 1151DP4PAN and 1151DB4PAN Transmitters, General Electric 184C5988G131 Master Trip Units, Rosemount 710DU Slave Trip Units and Agastat ETR14D3BC750 Time Delay Relays. The General Electric and Rosemount Trip Units are functionally checked and setpoint verified more frequently, and if necessary, recalibrated. These more frequent testing requirements remain unchanged. Therefore, an increase in the surveillance interval to accommodate a 24 month fuel cycle does not affect the Trip Units with respect to drift. The Rosemount Transmitters' and the Agastat Time Delay Relays' drift was determined by quantitative analysis. The drift value determined was used in the development of, confirmation of, or revision to the current plant setpoint and the Technical Specification Allowable Value. The results of this analysis support a 24 month surveillance interval.

TECHNICAL CHANGES - LESS RESTRICTIVE

LE.1 (cont'd) **Functional Unit 2**, Reactor Vessel Pressure - High

This function is performed by Rosemount 1151P9E22 Transmitters and Rosemount 710DU Master Trip Units. The Rosemount Master Trip Units are functionally checked and setpoint verified more frequently, and if necessary, recalibrated. These more frequent testing requirements remain unchanged. Therefore, an increase in the surveillance interval to accommodate a 24 month fuel cycle does not affect the Rosemount Master Trip Units with respect to drift. The Rosemount Transmitters' drift was determined by quantitative analysis. The drift value determined was used in the development of, confirmation of, or revision to the current plant setpoint and the Technical Specification Allowable Value. The results of this analysis support a 24 month surveillance interval.

Based on the design of the instrumentation and the drift evaluations, it is concluded that the impact, if any, on system availability is minimal as a result of the change in the surveillance test interval.

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) Trip Setpoints for the Improved Technical Specifications (ITS) Allowable Values. ITS Section 3.3 reflects Allowable Values consistent with the philosophy of BWR ISTS, NUREG-1433, 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LF.1 (cont'd) 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 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 CTS 3.2.C Actions 2, 4, 5 and 6 require the unit to be placed in Startup (Mode 2) within 6 hours if the ATWS-RPT instrumentation is not restored within the allowed out-of-service times. The purpose of the ATWS-RPT instrumentation is to trip the recirculation pumps. Therefore, an additional Required Action is proposed, ITS 3.3.4.1 Required Action D.1, to allow removal of the associated recirculation pump from service in lieu of being in MODE 2 within 6 hours. Since this action accomplishes the functional purpose of the ATWS-RPT instrumentation and enables continued operation in a previously approved condition, this change does not have a significant effect on safe operation. 8

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.2 CTS 3.2.C Action 3 requires the associated Trip System to be declared inoperable when two reactor vessel water level channels or two reactor vessel pressure channels in the same Trip System are inoperable in one or two trip systems. Declaring the Trip System inoperable would require restoration of the inoperable channels, as required by CTS 3.2.C Action 5 or 6. Placing the inoperable channels in trip is not allowed as an option. ITS 3.3.4.1 Required Action A.1 provides an option to place all inoperable channels in the tripped condition. This conservatively compensates for the inoperable status, restores the single failure capability and provides the required initiation capability of the instrumentation. Therefore, providing this option does not impact safety. However, if this action would result in system actuation, then declaring the system inoperable is the preferred action.

L.3 CTS 3.2.C Action 5 requires that when one Trip System is inoperable, 72 hours are provided to restore the Trip System. CTS 3.2.C Action 6 requires that when both Trip Systems are inoperable, 1 hour is provided to restore one Trip System. As described in CTS 3.2.C Action 3, a Trip System is inoperable when two channels of the same Function (i.e., reactor vessel water level or reactor vessel pressure) are inoperable in the Trip System. ITS 3.3.4.1 ACTION B addresses trip Function capability, not Trip System capability. A trip Function is maintained when sufficient channels are Operable or in trip, such that the ATWS-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. This requires two channels of the Function, in the same trip system, to each be Operable or in trip. The following is a description of the manner in which the ITS is applied, relative to the CTS.

a) When a single Trip System is inoperable under the CTS requirements, either due to two inoperable reactor vessel water level channels or two inoperable reactor vessel pressure channels, or both, the ITS will not have an inoperable Function. Therefore, ITS 3.3.4.1 ACTION A would apply, which allows 14 days to restore channels. This is consistent with the CTS 3.2.C Action 2 and Action 4 time. While in this condition, the ATWS-RPT System is still capable of tripping both recirculation pumps on either Function. In addition, two similar channels inoperable is functionally equivalent to one channel inoperable (which the CTS allows in Action 2) after the change described in Discussion of Change M.1 above; the Trip System will not provide a trip signal from the given Function.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd)

b) When both Trip Systems are inoperable under the CTS requirements due to two channels of the same Function being inoperable in both Trip Systems, 1 hour is allowed by CTS 3.2.C Action 6 to restore one of the Trip Systems (by restoring the channels in the Trip System). In the ITS, when two channels of the same Function are inoperable in both Trip Systems, one Function will be inoperable. Therefore, ITS 3.3.4.1 ACTION B would apply, which allows 72 hours to restore the inoperable channels. This is acceptable since while in this condition, the ATWS-RPT System is still capable of tripping both recirculation pumps on the other Function and operator action can still be taken to trip the recirculation pumps during this beyond design basis event. In CTS 3.2.C Action 3, this same condition requires entry into CTS 3.2.C Action 6 where only one hour is provided to restore one Trip System to Operable status.

c) When both Trip Systems are inoperable under the CTS requirements due to two channels of one Function being inoperable in one Trip System and two channels of the other Function being inoperable in the other Trip System, the ITS will not have an inoperable Function. Therefore, ITS ACTION A would apply, which allows 14 days to restore channels. In CTS 3.2.C Action 3, this same condition requires entry into CTS 3.2.C Action 6 where only one hour is provided to restore one Trip System to Operable status. This is acceptable since while in this condition, the ATWS-RPT System is still capable of tripping both recirculation pumps on either Function.

In addition, when one channel is inoperable, the associated Function (either Reactor Vessel Steam Dome Pressure — High or Reactor Vessel Water Level — Low Low) cannot actuate the Trip System, since both channels of a Function must trip to actuate the Trip System (i.e., each Trip System is a two-out-of-two logic for each Function). This condition (one channel inoperable) is covered by CTS 3.2.C Action 2 and ITS 3.3.4.1 ACTION A. Since each Trip System is a two-out-of-two logic for each Function, two channels of the same Function inoperable in a Trip System is functionally equivalent to that currently allowed by CTS 3.2.C Action 2 (i.e., one channel inoperable). That is, with both channels of the same Function inoperable in a Trip System, the associated Function cannot actuate the Trip System, identical to the results when one channel of the associated Function is inoperable in a Trip System. CTS 3.2.C Action 2 allows this condition (loss of one

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd) Function in one Trip System) to exist for 14 days. Therefore, allowing ITS 3.3.4.1 ACTION A to apply when both channels of a Function in a Trip System are inoperable is acceptable.

d) When both Trip Systems are inoperable under the CTS requirements due to all channels of both Functions inoperable in both Trip Systems, the ITS will have two inoperable Functions. Therefore, ITS 3.3.4.1 ACTION C would apply, which allows 1 hour to restore channels. This is consistent with the CTS Action 6 time.

RELOCATED SPECIFICATIONS

None

ECCS ACTUATION INSTRUMENTATION

ACTION

A.8 Insert ACTION 30

~~ACTION 30 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:~~

~~a. With one CHANNEL inoperable, place the inoperable CHANNEL in the tripped condition within one hour or declare the associated ECCS system(s) inoperable.~~

~~b. With more than one CHANNEL inoperable, declare the associated ECCS system(s) inoperable.~~

A.8 Insert ACTION 31

~~ACTION 31 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:~~

L.1 or declare associated ECCS inoperable

~~a. For ADS, declare the associated ADS TRIP SYSTEM inoperable.~~

~~b. For CS, LPCI or HPCI, declare the associated ECCS system(s) inoperable.~~

A.8

ACTION 32 - ACTION B

With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within ~~one hour~~ 24 hours — A.8

A.8 Insert ACTION 33

~~ACTION 33 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within one hour; restore the inoperable CHANNEL to OPERABLE status within 7 days or declare the associated ECCS system(s) inoperable.~~

ACTION 34 - ACTION C

With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, restore the inoperable CHANNEL to OPERABLE status within 8 hours 24 — A.8 or declare the associated ECCS system(s) inoperable.

A.8 ACTION H

A.8 Insert ACTION 35

~~ACTION 35 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place at least one inoperable CHANNEL in the tripped condition within one hour or declare the HPCI system inoperable.~~

A.9 moved to ITS 3.3.8.1

ACTION 36 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within one hour, or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.9.A or 3.9.B, as appropriate.

A.8 Insert ACTION 37

Insert ACTION 38

ADMINISTRATIVE (continued)

A.10 CTS Table 3.2.B-1 Action 35 requires placing the inoperable channel in trip when a HPCI Condensate Storage Tank Level—Low or a HPCI Suppression Chamber Water Level—High channel is inoperable. A new Required Action has been added, ITS 3.3.5.1 Required Action D.2.2, to allow the HPCI pump suction to be aligned to the suppression pool in lieu of tripping the channel, if a Condensate Storage Tank Level—Low or Suppression Pool Water Level—High channel is inoperable. Since this proposed action results in the same condition as if the channel were tripped (tripping one channel results in the suction being aligned to the suppression chamber), this change is considered administrative.

A.11 CTS Table 4.2.B-1 requires a CHANNEL FUNCTIONAL TEST (CFT) of Functional Unit 3.g, the HPCI Manual Initiation Function, every 18 months. CTS 4.2.B.2 and proposed SR 3.3.5.1.6 require a LOGIC SYSTEM FUNCTIONAL TEST (LSFT) every 18 months (changed to 24 months - see Discussion of Change LD.1 below). Since the LSFT is a complete test of the logic, including the Manual Initiation switches, there is no need to require a CFT. Therefore, ITS 3.3.5.1 only requires an LSFT, and this change is considered administrative.

A.12 CTS Table 4.2.B-1 requires both a CHANNEL FUNCTIONAL TEST and a CHANNEL CALIBRATION of Functional Unit 4.c, ADS Initiation Timer, and Functional Unit 4.d, ADS Low Low Level Timer, (ITS Table 3.3.5.1-1 Functions 4.c, 5.c, 4.f, and 5.f) to be performed every 18 months. Since the CFT is included in the CTS and ITS definition of CHANNEL CALIBRATION and the CFT and the CHANNEL CALIBRATION are performed at the same Frequency, the CFT has been deleted for these Functions. The CHANNEL CALIBRATION will include the required testing of the CFT, therefore, this change is considered administrative.

A.13 Not used. C

A.14 CTS Table 4.2.B-1 Functional Unit 3.e, HPCI Reactor Vessel Water Level — High (Trip), identifies the CHANNEL CHECK as NA. Proposed ITS Table 3.3.5.1-1 Function 3.c, will include a CHANNEL CHECK in accordance with SR 3.3.5.1.1, at a Frequency of 12 hours. This requirement is being added consistent with the requirements currently identified for CTS Functional Units 1.a, 2.a, 3.a, and 4.a, since each of these Functional Units are associated with the same level instrumentation. Although this change identifies an additional requirement and may be considered more restrictive, since it is consistent with the current plant procedures, it is considered administrative.

ADMINISTRATIVE (continued)

A.15 These changes to CTS 3/4.2.B are provided in the Dresden 2 and 3 ITS consistent with the Technical Specification Change Request submitted to the NRC for approval per ComEd letter PSLTR #00-0056, dated February 21, 2000. As such, these changes are considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 Eight additional Functions have been added to help ensure the automatic actuation function of the ECCS subsystems to ensure the design basis events can be satisfied. These Functions are included in ITS Table 3.3.5.1-1 as follows:

Function 1.e, Core Spray Pump Start - Time Delay Relay,
Function 2.d, Reactor Steam Dome Pressure - Low (Break Detection),
Function 2.e, LPCI Pump Start - Time Delay Relay for Pumps B and D,
Function 2.g, Recirculation Pump Differential Pressure-High (Break Detection),
Function 2.h, Recirculation Riser Differential Pressure-High (Break Detection),
Function 2.i, Recirculation Pump Differential Pressure Time Delay-Relay (Break Detection),
Function 2.j, Reactor Steam Dome Pressure Time Delay-Relay (Break Detection), and
Function 2.k, Recirculation Riser Differential Pressure Time Delay-Relay (Break Detection)

The proposed Allowable Values for these Functions were determined consistent with the setpoint methodology described in Discussion of Change LF.1 below. Appropriate ACTIONS and Surveillances (SR 3.3.5.1.2, SR 3.3.5.1.5 and SR 3.3.5.1.6, as applicable) have also been added. This is an additional restriction on plant operation necessary to help ensure the ECCS Instrumentation are maintained Operable. B

M.2 A maximum Allowable Value has been added for the CS Pump Discharge Flow—Low (Bypass) Function (CTS Table 3.2.B-1 Functional Unit 1.d; ITS Table 3.3.5.1-1 Function 1.d) to ensure the valves will close to provide assumed ECCS flow to the core. The new Allowable Value is based upon the most recent setpoint calculations. This is an additional restriction on plant operation.

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.3 CTS Table 4.2.B-1 requires a CHANNEL FUNCTIONAL TEST (CFT) of Functional Unit 3.d, Suppression Chamber Water Level - High every 92 days. The Table does not currently require a CHANNEL CALIBRATION. The channels associated with this Function include a level switch that must trip at the specified setpoint (Allowable Value, see Discussion of Change A.2). Therefore, the proposed test for OPERABILITY is a CHANNEL CALIBRATION (SR 3.3.5.1.5) at a Frequency of 24 months consistent with drift analysis assumptions in the plant setpoint methodology. B

M.4 Not used.

M.5 Not used.

M.6 Not used.

M.7 Not used. B

M.8 CTS Table 3.2.B-1 Functional Unit 3.e (ITS Table 3.3.5.1-1 Function 3.c), HPCI - Reactor Vessel Water Level - High, only requires one channel of this Function to be Operable. The purpose of this Function is to close the HPCI turbine stop valve and pump discharge valve (i.e., trip the HPCI turbine) to prevent overflow into the main steam lines. This Function is monitored by two differential pressure transmitters. The output signals from these transmitters are arranged in a two-out-of-two logic for this Function. In order for the HPCI System to trip on high reactor vessel water level, both signals are required. Therefore, ITS Table 3.3.5.1-1 for Function 3.c will require two OPERABLE channels of the Reactor Vessel Water Level - High Function. This change represents an additional restriction on plant operation necessary to ensure the OPERABILITY of the HPCI - Reactor Vessel Water Level - High Function.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The detail in CTS Table 3.2.B-1 Note (h) related to the reference point of the Trip Setpoint of the reactor vessel water level instrumentation and the detail for CTS Table 3.2.B-1 for Functional Unit 3.d (Suppression Chamber Water Level) that the Trip Setpoint is referenced above the bottom of the chamber are proposed to be relocated to the UFSAR. The reference value for the associated A

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) Allowable Values for Reactor Vessel Water Level Functions specified in ITS Table 3.3.5.1-1 is to "instrument zero," as discussed in Discussion of Change A.4. This detail is not necessary to ensure the OPERABILITY of the ECCS instrumentation. The requirements of ITS 3.3.5.1 and the associated Surveillances are adequate to ensure the ECCS instrumentation is maintained OPERABLE. Therefore, this relocated detail is not required to be in the ITS to provide adequate protection of public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

LA.2 The system design detail specified in CTS Table 3.2.B-1, footnote (i), is proposed to be relocated to the Bases. Details relating to system design (e.g., valves associated with isolation signals) are unnecessary in the LCO. This detail is not necessary to ensure the OPERABILITY of the ECCS Instrumentation. The requirements of ITS 3.3.5.1 and the associated Surveillance Requirements are adequate to ensure the ECCS instruments are maintained OPERABLE. Therefore, 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.

LD.1 The Frequency for performing the LOGIC SYSTEM FUNCTIONAL TEST (LSFT) of CTS 4.2.B.2 and the CHANNEL FUNCTIONAL TEST for the HPCI Manual Initiation and the ADS Initiation and Low Low Level Timer Functions specified in CTS Table 4.2.B-1 (changes made in Discussion of Changes A.11 and A.12 above) has been extended from 18 months to 24 months in proposed SR 3.3.5.1.6. This SR ensures that ECCS logic will function as designed to ensure proper response 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.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 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. ECCS systems are tested on a more frequent basis during the operating cycle in

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) accordance with CTS 4.2.B.1 (proposed SRs 3.3.5.1.1, 3.3.5.1.2, 3.3.5.1.3, and 3.3.5.1.4). These SRs will ensure that a significant portion of the ECCS circuitry is operating properly and will detect significant failures of this circuitry. The ECCS network including the actuating logic is designed to be single failure proof and therefore, is highly reliable. 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 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.

LE.1 The Frequencies for performing CHANNEL CALIBRATIONS of CTS Table 4.2.B-1 for Functional Units 1.a, 1.d, 2.a, 2.d, 3.a, 3.c, 3.e, 4.a, 4.c and 4.d have been extended from 18 months to 24 months in proposed SR 3.3.5.1.5. 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.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 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.

TECHNICAL CHANGES - LESS RESTRICTIVE

LE.1 (cont'd) Extending the SR Frequency is acceptable because the ECCS network along with the ECCS initiation logic is designed to be single failure proof and therefore is highly reliable. Furthermore, the impacted ECCS instrumentation has been evaluated based on make, manufacturer and model number to determine that the instrumentation's actual drift falls within the design allowance in the associated setpoint calculation. The following paragraphs, listed by CTS Functional Unit number, identify by make, manufacturer and model number the drift evaluations performed:

Functional Units 1.a, 2.a: CS/LPCI Reactor Vessel Water Level - Low Low

This function is performed by Rosemount 1153DB4PA Transmitters, General Electric 184C5988C Master Trip Units, and Rosemount 710DU Slave Trip Units. The General Electric and Rosemount Trip Units are functionally checked and setpoint verified more frequently, and if necessary, recalibrated. These more frequent testing requirements remain unchanged. Therefore, an increase in the surveillance interval to accommodate a 24 month fuel cycle does not affect the Trip Units with respect to drift. The Rosemount transmitters' drift was determined by quantitative analysis. The drift value determined was used in the development of, confirmation of, or revision to the current plant setpoint and the Technical Specification Allowable Value. The results of this analysis support a 24 month surveillance interval.

Functional Unit 1.d, 2.d: CS/LPCI Discharge Flow - Low (Bypass)

This function is performed by Rosemount 1153DB3 and 1153DB5 Transmitters and 710DU Master Trip Units. The Rosemount Trip Units are functionally checked and setpoint verified more frequently, and if necessary, recalibrated. These more frequent testing requirements remain unchanged. Therefore, an increase in the surveillance interval to accommodate a 24 month fuel cycle does not affect the Rosemount Trip Units with respect to drift. The Rosemount transmitters' drift was determined by quantitative analysis. The drift value determined was used in the development of, confirmation of, or revision to the current plant setpoint and the Technical Specification Allowable Value. The results of this analysis support a 24 month surveillance interval.

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 3.2.B-1 Action 32 (for Functional Units 1.c and 2.c in MODES 4 and 5) requires the channels to be placed in the tripped condition within 24 hours. If this Action is not performed the CTS does not provide default actions, such as immediately declare the associated ECCS subsystem(s) inoperable. Thus, CTS 3.0.C is required to be entered. However, since CTS 3.0.C is not applicable in MODES 4 and 5, 10 CFR 50.36(c)(2) requires that the licensee notify the NRC if required by 10 CFR 50.72, and a Licensee Event Report (LER) be submitted to the NRC as required by 10 CFR 50.73. In lieu of these two requirements, ITS 3.3.5.1 ACTION H will require the associated supported subsystems to be declared inoperable immediately. This would require the associated ECCS subsystems to be declared inoperable and the actions of CTS 3.5.B taken. Since these actions have been previously approved (as modified by the DOCs for ITS 3.5.2), this change is considered acceptable. C

RELOCATED SPECIFICATIONS

None

A.1

3.2 - LIMITING CONDITIONS FOR OPERATION

A. Isolation Actuation

LCO 3.3.6.1

The isolation actuation instrumentation CHANNEL(s) shown in Table 3.2.A-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the ~~Trip Setpoint~~ column.

A.6 Allowable Values

APPLICABILITY:

As shown in Table 3.2.A-1.

A.2 Add proposed ACTIONS Note

ACTION:

ACTIONS A and B

1. With an isolation actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the ~~Trip Setpoint~~ column of Table 3.2.A-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the ~~Trip Setpoint~~ value.

Allowable Values A.6

2. ~~With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] within one hour.~~

A.3 Insert CTS 3.2.A Action 2

~~a An inoperable CHANNEL need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.2.A-1 for that trip function shall be taken.~~

4.2 - SURVEILLANCE REQUIREMENTS

A. Isolation Actuation

Note 1 to Surveillance Requirements

1. Each isolation actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.A-1.

SR 3.3.6.1.7

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per ~~18~~ 24 months. 1C

LD.1

Table 3.3.6.1-1
TABLE 4.2.A-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Function Functional Unit	SR 3.3.6.1.1 CHANNEL CHECK	SR 3.3.6.1.2 SR 3.3.6.1.5 CHANNEL FUNCTIONAL TEST	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6 CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
2 1. PRIMARY CONTAINMENT ISOLATION		A.3 Q	24 months LE.1	
2.a a. Reactor Vessel Water Level - Low	S-1	M-2	E[a]-3 6	1, 2, 3
2.b b. Drywell Pressure - High(M) A.4	NA	M-2	Q-4	1, 2, 3
2.c c. Drywell Radiation - High	S-1	M-2	E-6	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c]	S	M	E[a]	1, 2, 3 & *
b. Drywell Pressure - High[b,c]	NA	M	Q	1, 2, 3
c. Reactor Building Ventilation Exhaust Radiation - High[c]	S	M	Q	1, 2, 3 & **
d. Refueling Floor Radiation - High[c]	S	M	Q	1, 2, 3 & **
1 3. MAIN STEAM LINE (MSL) ISOLATION		A.3 Q	24 months LE.1	
1.a a. Reactor Vessel Water Level - Low Low	S	M-2	E[a]-3 6	1, 2, 3
~~b. Deleted~~				
1.b, 1.c c. MSL Pressure - Low	NA	M-2	Q-4 M.4	1
1.d d. MSL Flow - High	S	M-2	E Q-4	1, 2, 3
1.e e. MSL Tunnel Temperature - High	NA	E-5 24 months LD.1	E-6	1, 2, 3

A.5 Moved to ITS 3.3.6.2

A.1

6 24 months for Function 1.c LE.1

Table 3.3.6.1-1

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	SR 3.3.6.1.1 CHANNEL CHECK	SR 3.3.6.1.2 SR 3.3.6.1.5 CHANNEL FUNCTIONAL TEST	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6 CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
5 **4. REACTOR WATER CLEANUP SYSTEM ISOLATION**				
5.a a. Standby Liquid Control System Initiation	NA	SR3.3.6.1.7 E 24 months LD.1 A.8	6 NA	1, 2, 3 L.2
5.b b. Reactor Vessel Water Level - Low	S-1	A.3 Q M-2	E-3 24 months LE.1	1, 2, 3
4 **5. ISOLATION CONDENSER**				
4.a a. Steam Flow - High	NA	M-2	Q-4	1, 2, 3
4.b b. Return Flow - High	NA	M-2	Q-4	1, 2, 3
3 **6. HIGH PRESSURE COOLANT INJECTION ISOLATION**				
3.a, 3.b a. Steam Flow - High	NA	M-2	6 E-3	1, 2, 3
3.c b. ~~Reactor Vessel~~ Steam Supply Line Pressure - Low	NA	M-2	E-3 -6	1, 2, 3
3.d c. Area Temperature - High	NA	E -5 24 months	E-6 24 months LE.1	1, 2, 3
7. SHUTDOWN COOLING ISOLATION		LD.1		
6.b a. Reactor Vessel Water Level - Low	S-1	M-2	E-3 -6	3, 4, 5
6.a b. Recirculation Line Water Temperature - High (Cut-in Permissive)	NA	M-2	E-6	1, 2, 3

A.1

Table 3.3.6.1-1
TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

A.5 Moved to ITS 3.3.6.2

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

92 A.3

SR 3.3.6.1.3
SR 3.3.6.1.6 (a) Trip units are calibrated at least once per ~~31~~ 92 days and transmitters are calibrated at the frequency identified in the table.

A.4

(b) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

A.5

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

Moved to ITS 3.3.6.2

~~(d) Deleted~~

ADMINISTRATIVE (continued)

A.5 The requirements identified in CTS Tables 3.2.A-1 and 4.2.A-1 related to Secondary Containment Isolation (including Notes (c), (d), (*), and (**) to Table 3.2.A-1 and Notes (b), (c), (*), and (**) to Table 4.2.A-1) have been moved to ITS 3.3.6.2, Secondary Containment Isolation Instrumentation. Any technical changes to these requirements are addressed in the Discussion of Changes for ITS 3.3.6.2.

A.6 CTS 3.2.A requires the isolation actuation instrumentation setpoints to be set consistent with the Trip Setpoint values shown in Table 3.2.A-1. CTS 3.2.A Action 1 requires the CHANNEL to be declared inoperable when the setpoint is less conservative than the value shown in the Trip Setpoint column of Table 3.2.A-1. Table 3.2.A-1 includes a "Trip Setpoint" column. It is proposed to re-label this column as "Allowable Value" consistent with the format of the BWR ISTS, NUREG-1433, Rev. 1 (ISTS Table 3.3.6.1-1). In accordance with current plant procedures and practices, the Trip Setpoints specified in CTS Table 3.2.A-1 are applied as the Operability limit for the associated instruments. Therefore, the use of the term "Trip Setpoint" in the CTS is the same as the use of the term "Allowable Value" in the ITS. This proposed change does not modify the actual trip setpoints specified in CTS Table 3.2.A-1 for the isolation actuation instrumentation Functions or the Allowable Values specified in ITS Table 3.3.6.1-1 (see Discussion of Change LF.1 below for proposed changes to the Trip Setpoints/Allowable Values). Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

A.7 An action to "declare the affected system inoperable," as presented in CTS Table 3.2.A-1 Action 23, is an unnecessary reminder that other Technical Specifications may be affected. This is essentially a "cross reference" between Technical Specifications that has been determined to be adequately provided through training. In addition, the definition of "OPERABILITY" in ITS Section 1.1 would also ensure that the affected systems rendered inoperable by isolation of an affected line are declared inoperable. Therefore, this deletion is administrative.

A.8 The CHANNEL FUNCTIONAL TEST (CFT) requirement for CTS Table 4.2.A-1 Functional Unit 4.a, Standby Liquid Control (SLC) System Initiation has been deleted. The CFT is redundant to the LOGIC SYSTEM FUNCTIONAL TEST (LSFT). The SLC System Initiation channels have no adjustable setpoints, but are based on switch manipulation. The LSFT (proposed SR 3.3.6.1.7), which applies to ITS Table 3.3.6.1-1 Function 5.a (SLC System Initiation), tests all contacts and will provide proper testing of the channels tested by a CFT. Therefore, this deletion is considered administrative. C

ADMINISTRATIVE (continued)

A.9 CTS Table 3.2.A-1 (Isolation Actuation Instrumentation) provides footnote "e" for Functional Unit 7.b (Recirculation Line Water Temperature - High) stating that "only one TRIP SYSTEM" is provided. The provisions of footnote "e" are not retained for proposed ITS Table 3.3.6.1-1, Function 6.a. The two required channels provide inputs to a single trip string which in turn provides input to two trip systems. Since this change does not change the number of OPERABLE channels required for the Function per trip system and a description of the logic is provided in the Bases, this change is considered administrative.

A.10 CTS 3.2.A and CTS Table 3.2.A-1 require Functional Unit 3.e, Main Steam Line (MSL) Tunnel Temperature—High, to have at least 2 channels (of the 4) in each of 2 sets OPERABLE per trip system. It is proposed to clarify this requirement by replacing the words "2 of 4 in each of 2 sets" with "2 per trip string" such that the requirement is consistent with the terminology used in BWR ISTS, NUREG-1433, Rev. 1, for describing other similar trip logic schemes. The MSL Tunnel Temperature—High Functional Unit includes a total of 16 temperature switches, four for each steam tunnel area. One channel from each steam tunnel area inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation. According to the CTS terminology, a "set" refers to the four area temperature switches that are arranged in a series contact scheme. Each "set" of four temperature switch contacts open on high temperature to actuate (de-energize) a logic relay. The BWR ISTS would refer to this trip logic scheme as a "trip string." Thus, the CTS terminology for a "set" is equivalent to the BWR ISTS terminology for a "trip string." Furthermore, since there are two trip strings per trip system, the minimum channel requirement of "2 of 4 in each of 2 sets" is equivalent to the proposed requirement of "2 per trip string." This change is considered a presentation preference change since it serves only to clarify an existing requirement by using the BWR ISTS terminology. As such, this change is administrative.

A.11 The Trip Setpoint for Functional Units 1.a, 4.b, and 7.a, Reactor Vessel Water Level - Low, and Functional Unit 3.a, Reactor Vessel Water Level-Low Low, in Table 3.2.A-1 is referenced to the top of active fuel. The reference value for the associated Allowable Values specified in ITS Table 3.3.6.1-1 is to "instrument zero." This change has been made for human factors considerations. The indications in the control room can be directly associated with the value in the ITS. Any changes to the Trip Setpoints are addressed in Discussion of Changes A.6 and LF.1, therefore this change is considered administrative. A

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 An Allowable Value for a Function has been added, ITS Table 3.3.6.1-1 Function 1.c. This Function is the Main Steam Line Low Pressure—Timer (or Time Delay). This Function is required to ensure the OPERABILITY of the current and proposed MSL Pressure—Low Function (CTS Table 3.2.A-1 Function 3.c and ITS Table 3.3.6.1-1 Function 1.b). This Function provides a time delay for the MSL Pressure—Low Function to ensure an inadvertent main steam line isolation does not occur during transients which result in reactor steam dome pressure pertubations. However, the delay is limited to ensure proper operation during pressure regulator failure event. The proposed Allowable Value was determined consistent with the methodology described in Discussion of Change LF.1 below. This change is an additional restriction on plant operation necessary to ensure the design basis accident analysis assumptions are satisfied.

M.2 The minimum required channels for the Standby Liquid Control System Initiation Function in CTS Table 3.2.A-1 (Functional Unit 4.a) is NA. For the same Function in the ITS (ITS Table 3.3.6.1-1 Function 5.a) the required channels per trip system is specified to be 1. The switch provides trip signal inputs to one trip system in any position other than "OFF." For this Specification, the SLC initiation switch is considered to provide 1 channel input into the trip system. Since the requirement is more explicit, this change is considered more restrictive on plant operations.

M.3 Not used.

M.4 The Frequency of the CHANNEL CALIBRATION requirement for CTS Table 4.2.A-1, Functional Unit 3.d, Main Steam Line Flow — High has been increased from 18 months to 92 days (proposed ITS SR 3.3.6.1.4). 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 Main Steam Line Flow — High Functional Unit is maintained OPERABLE.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details in CTS 3.2.A Action 2 footnote a, relating to placing channels in trip, are proposed to be relocated to the Bases. The ACTIONS of ITS 3.3.6.1 ensure inoperable channels are placed in trip (which effectively trips the trip system) or remedial actions are taken to compensate for the inoperability, as appropriate. As a result, these relocated details are not necessary for ensuring the appropriate actions are taken in the event of inoperable primary containment isolation instrumentation channels. As such, these 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.

LA.2 The detail CTS Table 3.2.A-1 Note (i) related to the reference setting of the level instrumentation is proposed to be relocated to the UFSAR. The reference value for the associated Allowable Values specified in ITS Table 3.3.6.1-1 is to "instrument zero," as discussed in Discussion of Change A.11. This detail is not necessary to ensure the OPERABILITY of the primary containment isolation instrumentation. The requirements of ITS 3.3.6.1 and the Surveillances are adequate to ensure the primary containment isolation instrumentation is maintained OPERABLE. Therefore, the relocated detail is 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. A

LA.3 The detail in CTS 3.2.A-1 Note (f) that the Standby Liquid Control System Initiation Function channel closes only reactor water cleanup system isolation valves is proposed to be relocated to the Bases. The requirement in proposed LCO 3.3.6.1 that the primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE, the listed Function for the SLC System Initiation in Table 3.3.6.1-1, and the proposed Surveillances will ensure this Function is maintained OPERABLE. Therefore, 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.

LD.1 The Frequency for performing the LOGIC SYSTEM FUNCTIONAL TEST (LSFT) of CTS 4.2.A.2 (proposed SR 3.3.6.1.7) and the CHANNEL FUNCTIONAL TEST (CFT) for the MSL Tunnel Temperature—High, SLC System Initiation (changed to LSFT in Discussion of Change A.8 above), and C

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) HPCI Area Temperature—High Functions specified in CTS Table 4.2.A-1 (proposed SR 3.3.6.1.5) has been extended from 18 months to 24 months. This SR ensures that Isolation Actuation Instrumentation logic will function as designed to ensure proper response 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.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 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. Most instrument channels are tested on a more frequent basis during the operating cycle in accordance with CTS 4.2.A.1, the CFT. This testing of the isolation instrumentation ensures that a significant portion of the Isolation Actuation Instrumentation circuitry is operating properly and will detect significant failures of this circuitry. The PCIVs including the actuating logic is designed to be single failure proof and therefore, is highly reliable. 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: 1C

> "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 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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.

LE.1 The Frequency for performing the CHANNEL CALIBRATION Surveillance of current Surveillance 4.2.A and Table 4.2.A-1 (proposed SR 3.3.6.1.6) has been extended from 92 days (for the Main Steam Line Pressure - Timer) and 18 months (for all other Functional Units listed below) to 24 months. The proposed change will allow this Surveillance to extend the Surveillance Frequency 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 SR 3.0.2). The subject SR ensures that the Isolation instruments will function as designed during an analyzed event. Extending the SR Frequency is acceptable because the Primary Containment Isolation System along with the Isolation initiation logic is designed to be single failure proof and, therefore, is highly reliable. Furthermore, the impacted Isolation instrumentation has been evaluated based on make, manufacturer and model number to determine that the instrumentation's actual drift falls within the design allowance in the associated setpoint calculation. The following paragraphs, listed by CTS Functional Unit number, identify by make, manufacturer and model number the drift evaluations performed:

Functional Unit 1.a: Reactor Vessel Water Level - Low

This function is performed by Rosemount 1153DB4PAN Transmitters and 710DU Master and Slave Trip Units. The Rosemount Trip Units are functionally checked and setpoint verified more frequently, and if necessary, recalibrated. These more frequent testing requirements remain unchanged. Therefore, an increase in the surveillance interval to accommodate a 24 month fuel cycle does not affect the Rosemount Trip Units with respect to drift. The Rosemount Transmitters' drift was determined by quantitative analysis. The drift value determined was used in the development of, confirmation of, or revision to the current plant setpoint and the Technical Specification Allowable Value. The results of this analysis support a 24 month surveillance interval.

A.1

LOP Instrumentation A.2

INSTRUMENTATION

Table 3.3.8.1-1

~~ECCS Actuation~~ 3/4.2.B

TABLE 3.2.B-1 (Continued)

A.2 LOP

~~ECCS ACTUATION~~ INSTRUMENTATION

A.2

Insert CTS Table 3.2.B-1 Note (a) A.6

TABLE NOTATION

M.3

(a) ~~A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the associated Functional Unit maintains ECCS initiation capability.~~

(b) Also actuates the associated emergency diesel generator.

(c) When the system is required to be OPERABLE per Specification 3.5.B.

(d) Not required to be OPERABLE when reactor steam dome pressure is ≤150 psig. See ITS 3.3.5.1

Applicability (e) Required when the associated diesel generator is required to be OPERABLE per Specification 3.9.B.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

Function 2.b (g) With no LOCA signal present, there is an additional time delay of 5 ± 0.25 minutes. LF.1

(h) Reactor water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero). See ITS 3.3.5.1

(i) Provides signal to pump suction valves only.

Function 2.a (j) There is an inherent time delay of 7 ± 1.4 seconds on degraded voltage. LF.1

A.6 Insert CTS Table 3.2.B-1 Note(a)

(a) When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed as follows: See ITS 3.3.5.1

1) For up to six hours for Functional Units 3.e, 3.f, and 3.g; and

2) For up to ~~six~~ 2 hours for Functional Units other than 3.e, 3.f, and 3.g provided the functional unit maintains LOP actuation capability.

A.2

M.3

C

Note 2 to Surveillance Requirements

ADMINISTRATIVE

A.4 (cont'd) inoperable channel is allowed a certain time to complete the Required Actions. Since this change only provides more explicit direction of the current interpretation of the existing specifications, this change is considered administrative.

A.5 CTS Table 3.2.B-1 ACTION 36 requires the DG to be declared inoperable and to take the ACTION required by Specification 3.9.A or 3.9.B, as appropriate, when the inoperable LOP instrumentation channel is not tripped within 1 hour. The format of the ITS does not include providing "cross references." ITS 3.8.1 and ITS 3.8.2 adequately prescribe the Required Actions for an inoperable DG without such references. Therefore, the existing reference in CTS Table 3.2.B-1 ACTION 36 to "take the ACTION required by Specification 3.9.A or 3.9.B" serves no functional purpose, and its removal is purely an administrative difference in presentation.

A.6 This change to CTS Table 3.2.B-1 is provided in the Dresden ITS consistent with the Technical Specifications Change Request submitted to the NRC for approval per ComEd letter JMHLTR 00-0002, dated January 11, 2000. As such, these changes are considered to be administrative. C

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS Tables 3.2.B-1 and 4.2.B-1 require the LOP instruments to be OPERABLE during MODES 4 and 5 only when the associated DG is required to be OPERABLE (as stated in footnote (e) to Table 3.2.B-1 and footnote (c) to Table 4.2.B-1). The Applicability is being changed to be when the associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources — Shutdown," which in ITS 3.3.8.1 requires the LOP instrumentation to be OPERABLE not only during MODES 4 and 5, but also during movement of irradiated fuel assemblies in the secondary containment (which could be when the unit is defueled). This will ensure the DGs can be properly actuated at all times when they are required to be OPERABLE and is an additional restriction on plant operation.

M.2 A new Allowable Value has been added for the LOP Function. The maximum Allowable Value has been added for CTS Table 3.2.B-1 Degraded Voltage Function (ITS Table 3.3.8.1-1 Function 2.a) to prevent inadvertent power supply transfer. The new maximum Allowable Value represents an additional restriction on plant operation. B

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.3 CTS Table 3.2.B-1 Note(a) allows a Loss of Power Instrumentation channel to be inoperable to perform required Surveillances and not enter the required Actions for 6 hours, provided the Function Unit maintains actuation capability. ITS 3.3.8.1 Surveillances Note 2 will only allow this exception for 2 hours. This change is more restrictive on plant operations and is consistent with NUREG-1433, Rev. 1. C

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The detail in CTS Table 3.2.B-1 Functional Unit 6.a relating to the methods (on decreasing voltage) for determining the 4160 V ESS Bus Undervoltage (Loss of Voltage) Setpoint is proposed to be relocated to the Bases. This detail is not necessary to ensure the OPERABILITY of the loss of power instrumentation. The requirements of ITS 3.3.8.1 and proposed SR 3.3.8.1.2 are adequate to ensure the loss of power instruments are maintained OPERABLE. Therefore, 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.

LD.1 The Frequency for performing the LOGIC SYSTEM FUNCTIONAL TEST of CTS 4.2.B.2 and the CHANNEL FUNCTIONAL TEST of CTS Table 4.2.B-1 for Functional Unit 5.a, 4.16 kV Emergency Bus Undervoltage (Loss of Voltage) have been extended from 18 months to 24 months in proposed SR 3.3.8.1.3 and SR 3.3.8.1.5. B These SRs ensure that LOP Instrumentation logic will function as designed to ensure proper response 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.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 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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 LOP instrumentation including the actuating logic is designed to be single failure proof and therefore, is highly reliable. 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 inherent system and component reliability, the impact, if any, on system availability is minimal as a result of the change in the surveillance test interval. 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.

LE.1 The Frequency for performing the CHANNEL CALIBRATION of CTS 4.2.B.1 (Functional Unit 5.a) has been extended from 18 months to 24 months in proposed SR 3.3.8.1.4. This SR ensures that LOP Instrumentation associated with the 4.16 kV Emergency Bus Undervoltage - Loss of Voltage channels will function as designed to ensure proper response 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.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 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. B

TECHNICAL CHANGES - LESS RESTRICTIVE

LE.1 (cont'd) Extending the SR Frequency is acceptable because the electrical power sources are designed to be single failure proof and therefore are highly reliable. Major deviations in the circuitry will be discovered during the cycle since the CHANNEL FUNCTIONAL TEST of both the loss of voltage instrumentation and the time delay relays are performed more frequently. Furthermore, the impacted LOP instrumentation has been evaluated based on make, manufacturer and model number to determine that the instrumentation's actual drift falls within the design allowance in the associated setpoint calculation.

This function is performed by General Electric 12IAV69A1A relays. The GE relays' drift was determined by quantitative analysis. The drift value determined was used in the development of, confirmation of, or revision to the current plant setpoint and the Technical Specification Allowable Value. The results of this analysis support a 24 month surveillance interval.

Based on the design of the instrumentation and the drift evaluations, it is concluded that the impact, if any, on system availability is minimal as a result of the change in the surveillance test interval.

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) Trip Setpoints for the Improved Technical Specifications (ITS) Allowable Values. ITS Section 3.3 reflects Allowable Values consistent with the philosophy of BWR ISTS, NUREG-1433, 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LF.1 (cont'd) 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 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"

None

RELOCATED SPECIFICATIONS

None

<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

-----NOTES----- 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.	12 hours
	OR	
	A.2 Place associated trip system in trip.	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.	6 hours
	OR	
	B.2 Place one trip system in trip.	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 Function 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.

C

6. ISTS SR 3.3.1.1.14 has been deleted since the APRM Flow Biased Neutron Flux—High circuit does not include the simulated heat flux time constant. However, a new SR has been added to the APRM Flow Biased Neutron Flux—High Function to perform a CHANNEL CALIBRATION of the flow converters (ITS SR 3.3.1.1.17). In addition, a Note to ISTS SR 3.3.1.1.11 (ITS SR 3.3.1.1.15) is added to clarify the applicability of the CHANNEL CALIBRATION SRs to the flow converters. Subsequent SRs have been renumbered, as required.

7. The bracketed requirement has been deleted since it does not apply to the current Dresden 2 and 3 licensing basis. Subsequent Functions have been renumbered, as applicable.

8. The ITS SR 3.3.1.1.1, CHANNEL CHECK, cannot be performed, since no indicators are provided, for the channels associated with the following Functions. Therefore, the CHANNEL CHECK requirement has been deleted from the associated Function Surveillance Requirements in ITS Table 3.3.1.1-1.

   Function 3, Reactor Vessel Steam Dome Pressure — High
   Function 6, Drywell Pressure — High
   Function 7, Scram Discharge Volume Water Level — High

   This is consistent with the current licensing basis.

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

10. ITS Table 3.3.1.1-1 Function 10, Turbine Condenser Vacuum—Low, and associated footnote (c) have been added consistent with the current licensing basis for RPS Instrumentation. Subsequent Functions have been renumbered, as required. In addition, the Turbine Condenser Vacuum — Low (ITS Table 3.3.1.1-1 Function 10) Function is required to be calibrated every 31 days in accordance with the current setpoint methodology. Therefore, an SR has been added (ITS SR 3.3.1.1.10) to ensure the licensing basis is retained. Subsequent SRs have been renumbered, as required.

11. The Frequency for ISTS SR 3.3.1.1.6 has been changed from "Prior to withdrawing SRMs from the fully inserted position" to "Prior to fully withdrawing SRMs." The current licensing basis for Dresden 2 and 3 only requires the SRM/IRM overlap to be verified during a reactor startup. It does not require the overlap verification prior to withdrawing the SRMs from the fully inserted position. While the current practice of c

11. (continued)

Dresden 2 and 3 is to maintain the SRMs inserted until SRM/IRM overlap is verified, withdrawing the SRMs prior to the IRMs coming on range will reduce the burnup of the SRMs. In addition, the current LaSalle 1 and 2 practice is to partially withdraw the SRMs prior to verifying overlap. Therefore, ITS SR 3.3.1.1.6 has been modified to be consistent with the current LaSalle 1 and 2 practice, and is consistent with current licensing basis.

12. A requirement to perform an RPS RESPONSE TIME test on the Drywell Pressure—High Function channels has been added since the Function is credited in the safety analyses.

13. ITS Required Action F.2 for Function 5, Main Steam Isolation Valve - Closure, and Function 10, Turbine Condenser Vacuum - Low, has been added to require reducing reactor pressure to < 600 psig. The Applicability of ITS Table 3.3.1.1-1 Function 5 has also been revised to include MODE 2 and footnote (c), i.e., MODE 2 with reactor pressure $\geq$ 600 psig. These changes are consistent with the current licensing basis for RPS Instrumentation.

14. Typographical error corrected.

15. An Actions Note is added to allow time to adjust the gain for the APRMs. This Note is included in CTS Table 4.1.A-1 as Note (d), and is based on both the time frame necessary to accomplish multiple channel gain adjustments and the impact on safety. Only two hours are provided if the GAF is non-conservative; but 12 hours are provided if the GAF is out of limits low since this makes the trip setpoint conservative.

16. TSTF-264 deletes the Surveillances for SRM/IRM overlap during startup and the APRM/IRM overlap during shutdown. The TSTF states that these SRs are unnecessary since they duplicate the requirements of the Channel Check. However, the Channel Check definition does not specifically require overlap checks. There are other instruments that have overlapping ranges (e.g., reactor water level instruments), and no "overlap" checks are implied by the Channel Check requirements for these instruments. Also, as stated in the TSTF Bases portion of the change, the SRM/IRM overlap check is only applicable during a startup and the APRM/IRM overlap check is only required during a shutdown. It would appear that if the Channel Check definition requires overlap checks, it would require the checks both during a startup and during a

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.3.2.2 - FEEDWATER SYSTEM AND MAIN TURBINE HIGH WATER LEVEL TRIP INSTRUMENTATION

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

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

3. The Dresden 2 and 3 Feedwater System and Main Turbine High Water Level Trip Instrumentation includes four channels. ISTS 3.3.2.2 ACTIONS A and B are written for a three channel design. For a three channel design, when two of the three channels are inoperable, a loss of function has occurred. However, the Dresden 2 and 3 design is such that with two channels inoperable, a loss of function may not have occurred. Therefore, ISTS 3.3.2.2 Condition A has been modified to be applicable to one or more inoperable channels, and ISTS 3.3.2.2 Condition B has been modified to be applicable to when a loss of function has occurred (i.e., trip capability not maintained). This change is consistent with the intent of the ISTS, which requires the 2 hour Completion Time of ACTION B to be applicable when a loss of function has occurred.

4. ISTS 3.3.2.2 Required Action C.1 (ITS Required Action C.2) requires a reduction in Thermal Power to ≤ 25% RTP if the Feedwater System and Main Turbine High Water Level Trip Instrumentation is not restored to Operable status. The instrumentation indirectly supports maintaining MCPR above limits during a feedwater controller failure, maximum demand event. This is accomplished by tripping the main turbine, with the main turbine trip resulting in a subsequent reactor scram. When the instrumentation is inoperable solely due to an inoperable feedwater pump breaker, the unit can continue to operate with the feedwater pump removed from service (Dresden 2 and 3 have three 50% capacity feedwater pumps). Therefore, an additional Required Action is proposed, ITS 3.3.2.2, Required Action C.1, to allow removal of the associated feedwater pump(s) from service in lieu of reducing Thermal Power. This Required Action will only be used if the instrumentation is inoperable solely due to an inoperable feedwater pump breaker, as stated in the Note to ITS 3.3.2.2 Required Action C.1. Since this Required Action accomplishes the functional purpose of the Feedwater System and Main Turbine High Water Level Trip Instrumentation, enables continued operation in a previously approved condition, and still supports maintaining MCPR above limits (since the reactor scram is the result of a turbine trip signal, which is not impacted by this change), this change does not have a significant effect on safe operation. This change is also consistent with TSTF-297. In addition, ISTS 3.3.2.2 Required Action C.1 has been renumbered due to this addition. | C

5. Not used.

8. ISTS Table 3.3.5.1-1 Function 2.e requires a minimum time for the ECCS pump start time delay relays. The ISTS Bases states that the minimum time is to ensure that excess loading will not cause failure of the power source; i.e., the minimum Allowable Value is chosen to be long enough so that most of the starting transient of the first pump is complete before starting the second pump on the same 4.16 kV emergency bus. Failure of this portion of the instrumentation will result in the DG being inoperable; it does not necessarily result in the inoperability of the ECCS pump. The ECCS analysis assumes the pumps are operating at a certain time; starting the pumps sooner than assumed does not invalidate the ECCS analysis. This requirement is adequately covered by ITS SR 3.8.1.18, which requires the interval between each sequenced load block to be within $\pm$ 10% of the design interval for each load sequence time delay relay. The ITS Bases for this SR states that it ensures that a sufficient time interval exists for the DG to restore frequency and voltage prior to applying the next load and that safety analyses assumptions regarding ESF equipment time delays are not violated. Therefore, if a time delay relay actuated too soon such that a power source was affected, the requirements of SR 3.8.1.18 would not be met and the affected DG or offsite circuit would be declared inoperable and the ACTIONS of ITS 3.8.1 taken. Therefore, there is no reason to require minimum times in the ECCS Instrumentation Specification. This is also consistent with current licensing basis, which does not have minimum time requirements for the ECCS pump start time delay relays in the ECCS Instrumentation Specification.

9. The current Dresden 2 and 3 design does not include the ADS Reactor Vessel Water Level—Low, Level 3 (Confirmatory) Function (ISTS Functions 4.d and 5.d). Therefore, these Functions have been deleted and the remaining Functions have been renumbered, where applicable, to reflect these deletions.

10. ISTS Table 3.3.5.1-1 Function 2.e, Reactor Vessel Shroud Level—Level 0, has been relocated as documented in the Discussion of Changes for CTS 3/4.2.I. Subsequent Functions have been renumbered as required. C

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

<CTS>

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation Function.

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

	SURVEILLANCE	FREQUENCY
<T4.2.A-1>	SR 3.3.6.1.1 Perform CHANNEL CHECK.	12 hours
<T4.2.A-1>	SR 3.3.6.1.2 Perform CHANNEL FUNCTIONAL TEST.	[92] days
<T4.2.A-1>	[SR 3.3.6.1.3 Calibrate the trip unit.	[92] days]
<T4.2.A-1>	SR 3.3.6.1.4 Perform CHANNEL CALIBRATION.	92 days
	[SR 3.3.6.1.5 Perform CHANNEL FUNCTIONAL TEST.	~~[184] days~~ 24 Months]
<T4.2.A-1>	SR 3.3.6.1.6 Perform CHANNEL CALIBRATION.	~~[18]~~ 24 months
<4.2.A.2>	SR 3.3.6.1.7 Perform LOGIC SYSTEM FUNCTIONAL TEST.	~~[18]~~ 24 months

(continued)

5 5 11 5 C C C

<CTS>

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

Table 3.3.6.1-1 (page 1 of 6)
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]~~ 2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[~~SR 3.3.6.1.3~~]~~ SR 3.3.6.1.6 SR 3.3.6.1.7 ~~[SR 3.3.6.1.8]~~	≥ ~~[-113]~~ -56.77 inches
b. Main Steam Line Pressure — Low	1	~~[2]~~ 2	E	~~[SR 3.3.6.1.1]~~ ~~[~~SR 3.3.6.1.2~~]~~ SR 3.3.6.1.4 SR 3.3.6.1.7 ~~[SR 3.3.6.1.8]~~	≥ ~~[825]~~ 831 psig
d. ~~c.~~ Main Steam Line Flow — High	1,2,3	~~[2]~~ 2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ ~~[SR 3.3.6.1.4]~~ SR 3.3.6.1.7 ~~[SR 3.3.6.1.8]~~ SR 3.3.6.1.4	≤ ~~[138]% rated steam flow~~ 160.5 psid (Unit 2) ≤ 117.1 psid (Unit 3)
~~d. Condenser Vacuum — Low~~	~~1, 2(a), 3(a)~~	~~[2]~~	~~D~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7~~	~~≥ [7] inches Hg vacuum~~
e. Main Steam ~~Tunnel~~ Line Temperature — High	1,2,3	~~[8]~~ 2 per trip string	D	~~[SR 3.3.6.1.1]~~ SR 3.3.6.1.~~2~~ 5 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.6 SR 3.3.6.1.7 ~~[SR 3.3.6.1.8]~~	≤ ~~[194]~~ 200°F
~~f. Main Steam Tunnel Differential Temperature — High~~	~~1,2,3~~	~~[2]~~	~~D~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≤ [ ]°F~~
~~g. Turbine Building Area Temperature — High~~	~~1,2,3~~	~~[32]~~	~~D~~	~~[SR 3.3.6.1.1] SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≤ [200]°F~~
~~h. Manual Initiation~~	~~1,2,3~~	~~[1]~~	~~G~~	~~SR 3.3.6.1.7~~	~~NA~~
c. Main Steam Line Pressure - Timer	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 0.280 seconds (unit 2) ≤ 0.236 seconds (unit 3)

(continued)

~~(a) With any turbine [stop valve] not closed.~~

<CTS>

<T3.2.A-1>
<DOC M.3>
<T4.2.A-1>
<T3.2.A-1 Footnote (h)>

Table 3.3.6.1-1 (page 2 of 6)
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
2. Primary Containment Isolation					
a. Reactor Vessel Water Level — Low, ~~Level 3~~ 7	1,2,3	~~[2]~~	~~H~~ 3 G	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ 5 SR 3.3.6.1.6 SR 3.3.6.1.7 ~~[SR 3.3.6.1.8]~~ 6	≥ ~~[10]~~ 10.24 5 inches
b. Drywell Pressure — High	1,2,3	5 ~~[2]~~	~~H~~	9 ~~[SR 3.3.6.1.1]~~ SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ 8 ~~SR 3.3.6.1.6~~ Add SR 3.3.6.1.4 10 SR 3.3.6.1.7 ~~[SR 3.3.6.1.8]~~ 6	~~≤ [1.92]~~ psig ≤ 1.81 5
c. Drywell Radiation — High	1,2,3	~~[1]~~	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ ~~[138]~~ 77 5 R/hr
~~d. Reactor Building Exhaust Radiation — High~~	~~1,2,3~~	~~[2]~~	~~H~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8~~	~~≤ [60] mR/hr~~ 3
~~e. Refueling Floor Exhaust Radiation — High~~	~~1,2,3~~	~~[2]~~	~~H~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8~~	~~≤ [20] mR/hr~~
~~f. Manual Initiation~~	~~1,2,3~~	~~[1 per group]~~	~~G~~	~~SR 3.3.6.1.7~~	~~NA~~
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow — High	1,2,3	5 ~~[1]~~	F	9 ~~[SR 3.3.6.1.1]~~ SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ 5 SR 3.3.6.1.6 SR 3.3.6.1.7 6 ~~[SR 3.3.6.1.8]~~	~~≤ [303]% rated steam flow~~ 5 ≤ 290.16% of rated steam flow (Unit 2) ≤ 288.23% of rated steam flow (Unit 3)

3 Insert Function 3.b

(continued)

A
C
A
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A
C

<CTS>
<DOC M.3>
<T3.2.A-1 Footnote (h)>

3

Insert Function 3.b

b.	HPCI Steam Line Flow-Timer	1, 2, 3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≥ 3.2 seconds and ≤ 8.8 seconds

B C

<CTS>

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

Table 3.3.6.1-1 (page 3 of 6)
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
3. HPCI System Isolation (continued)					
c ~~b~~. HPCI Steam Supply Line Pressure — Low	1,2,3	~~[2]~~ 5	F	~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 ~~[~~SR 3.3.6.1.3~~]~~ SR 3.3.6.1.6 SR 3.3.6.1.7 ~~SR 3.3.6.1.8~~	≥ ~~[100]~~ 104 psig
~~c. HPCI Turbine Exhaust Diaphragm Pressure — High~~	~~1,2,3~~	~~[2]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≤ [20] psig~~
~~d. Drywell Pressure — High~~	~~1,2,3~~	~~[1]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.6 SR 3.3.6.1.7 [SR 3.3.6.1.8]~~	~~≤ [1.92] psig~~
d ~~e~~. HPCI ~~Pipe Penetration~~ ~~Room~~ Turbine Area Temperature — High	1,2,3	~~[2]~~ 4(a)	F	~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 ~~[SR 3.3.6.1.5]~~ SR 3.3.6.1.6 SR 3.3.6.1.7	≤ ~~[169]~~ 189 °F
~~f. Suppression Pool Area Ambient Temperature — High~~	~~1,2,3~~	~~[1]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≤ [169]°F~~
~~g. Suppression Pool Area Temperature — Time Delay Relays~~	~~1,2,3~~	~~[1]~~	~~F~~	~~SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≥ [NA] [minutes]~~
~~h. Suppression Pool Area Differential Temperature — High~~	~~1,2,3~~	~~[1]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≤ [42]°F~~
~~i. Emergency Area Cooler Temperature — High~~	~~1,2,3~~	~~[1]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≤ [169]°F~~
~~[j. Manual Initiation~~	~~1,2,3~~	~~[1 per group]~~	~~G~~	~~SR 3.3.6.1.7~~	~~NA]~~

(continued)

(a) All four channels must be associated with a single trip string.

Insert Function 4

4. Isolation Condenser System Isolation					
a. Steam Flow-High	1, 2, 3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 290.76% of rated steam flow
b. Return Flow-High	1, 2, 3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 30.2 inches water (Unit 2) ≤ 13.7 inches water (Unit 3)

<CTS>

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

Table 3.3.6.1-1 (page 5 of 6)
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
4. RCIC System Isolation (continued)					
~~f. RCIC Equipment Room Temperature — High~~	~~1,2,3~~	~~[1]~~	~~F~~	~~[SR 3.3.6.1.1] SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.4 SR 3.3.6.1.7~~	~~≤ [ ]°F~~
~~j. RCIC Equipment Room Differential Temperature — High~~	~~1,2,3~~	~~[1]~~	~~F~~	~~[SR 3.3.6.1.1] SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.4 SR 3.3.6.1.7~~	~~≤ [ ]°F~~
~~k. Manual Initiation~~	~~1,2,3~~	~~[1 per group]~~	~~G~~	~~SR 3.3.6.1.7~~	~~NA~~
5. Reactor Water Cleanup (RWCU) System Isolation					
~~a. Differential Flow — High~~	~~1,2,3~~	~~[1]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8~~	~~≤ [79] gpm~~
~~b. Area Temperature — High~~	~~1,2,3~~	~~[3] [1 per room]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.6 SR 3.3.6.1.7 [SR 3.3.6.1.8]~~	~~≤ [150]°F~~
~~c. Area Ventilation Differential Temperature — High~~	~~1,2,3~~	~~[3] [1 per room]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.6 SR 3.3.6.1.7 [SR 3.3.6.1.8]~~	~~≤ [67]°F~~
a. ~~d.~~ SLC System Initiation	1,2	1 ~~[2]~~ ~~(b)~~	~~I~~ H	SR 3.3.6.1.7	NA
b. ~~e.~~ Reactor Vessel Water Level — Low ~~Low,~~ ~~Level 2~~	1,2,3	~~[2]~~	F	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.6 SR 3.3.6.1.7 ~~[SR 3.3.6.1.8]~~	≥ ~~[-47]~~ 10.24 inches
~~f. Manual Initiation~~	~~1,2,3~~	~~[1 per group]~~	~~G~~	~~SR 3.3.6.1.7~~	~~NA~~

~~(b) SLC System Initiation only inputs into one of the two trip systems.~~

(continued)

<CTS>

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

Table 3.3.6.1-1 (page 6 of 6)
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
6. Shutdown Cooling System Isolation					
a. ~~Reactor Steam Dome Pressure – High~~ Recirculation Line Water Temperature – High 7	1,2,3	~~(1)~~ 2 5	F	9 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 ~~SR 3.3.6.1.3~~ 8 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ ~~[145] psig~~ 346°F 5
b. Reactor Vessel Water Level – Low~~, Level 3~~	3,4,5	~~[2]~~ (b) 1	~~J~~ I 3	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~SR 3.3.6.1.3~~ 5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ ~~[10]~~ 10.24 inches 5

A C A C

5 (b) ~~(c)~~ ~~Only one trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.~~

In MODES 4 and 5, provided Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required. 12

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.3.6.1 - PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

1. The proper Primary Containment Isolation Functions that are common to the RPS Instrumentation have been provided.

2. The Completion Time of ISTS 3.3.6.1 Required Action E.1 has been extended by 2 hours consistent with the current licensing basis.

3. Four new Primary Containment Isolation Functions have been added (ITS Table 3.3.6.1-1 Functions 1.c, 3.b, 4.a, and 4.b), consistent with current Dresden 2 and 3 Licensing Basis. In addition, 29 Functions have been deleted (ISTS Table 3.3.6.1-1 Functions 1.d, 1.f, 1.g, 1.k, 2.d, 2.e, 2.f, 3.c, 3.d, 3.f, 3.g, 3.h, 3.i, 3.j, 4.a, 4.b, 4.c, 4.d, 4.e, 4.f, 4.g, 4.h, 4.i, 4.j, 4.k, 5.a, 5.b, 5.c and 5.f) since they are not applicable to Dresden 2 and 3. The Functions and ACTIONS have been revised where applicable, to reflect these additions and deletions.

4. Editorial change made to be consistent with other similar requirements in the ITS or for clarity.

5. The brackets have been removed and the proper plant specific information/value has been provided. Table footnotes have been renumbered, as required.

6. ISTS SR 3.3.6.1.8, the Isolation System Response Time test, is not included in the Dresden ITS. This allowance is consistent with the current licensing basis reflected in the CTS. In addition, the Reviewer's Note has been deleted. The Note is not meant to be retained in the final version of the plant specific submittal.

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

8. The bracketed Surveillances have been deleted since they do not apply to the associated Function. These changes are consistent with the current licensing basis.

9. These Surveillances have been deleted since they can not be performed on the associated Function.

10. This additional Surveillance, requiring performance of a CHANNEL CALIBRATION once per 92 days, has been added consistent with the current setpoint calibration methodology (SR 3.3.6.1.4). As a result, ISTS SR 3.3.6.1.6 is deleted from the Table 3.3.6.1-1 Surveillance Requirement column, for the applicable Functions, for the same reason.

11. The CFT Surveillance associated with ITS 3.3.6.1-1 Function 1.d (Main Steam Tunnel Temperature—High) and Function 3.d (HPCI Turbine Area Temperature—High) has been revised to reflect current licensing requirements, therefore, SR 3.3.6.1.2 has been replaced with SR 3.3.6.1.5.

12. ISTS Table 3.3.6.1-1 footnote c (ITS footnote b) has been revised to reflect the specific design of the Shutdown Cooling System suction isolation valve logic.

13. The Main Steam Line Flow — High (ITS Table 3.3.6.1-1 Function 1.c) Function is required to have a CHANNEL CALIBRATION performed every 92 days in accordance with current setpoint methodology. Therefore, ITS SR 3.3.6.1.4 has been added to ensure that the Main Steam Line Flow — High Functional Unit is maintained OPERABLE. As a result, ISTS SR 3.3.6.1.6 (18 month CHANNEL CALIBRATION) has been removed from the Main Steam Line Flow — High Function since it is redundant to the added Surveillance Requirement.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve—Closure (continued)

stage pressure; therefore, ~~to consider this Function OPERABLE,~~ OPENING the turbine bypass valves ~~must remain shut at THERMAL POWER ≥ 30% RTP~~ may affect the OPERABILITY of this Function. B C

The Turbine Stop Valve—Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

TSTF-231

Eight channels of Turbine Stop Valve—Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function ~~if any three TSVs should close~~ even if one TSV should fail to close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ ~~30~~ 45% RTP. This Function is not required when THERMAL POWER is < ~~30~~% RTP since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Fixed Neutron Flux—High Functions are adequate to maintain the necessary safety margins. C 3

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function is the primary scram signal for the generator load rejection event analyzed in Reference ~~4~~ 12. For this event, the reactor scram reduces the amount of energy required to be absorbed and ~~, along with the actions of the EOC-RPT System,~~ ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure—Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure ~~transmitter~~ switch is associated with each control valve, and the signal from each ~~transmitter~~ is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ ~~30~~ 45% RTP. This is normally accomplished 3

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low (continued)

switches

automatically by pressure ~~transmitters~~ sensing turbine first ~~stage pressure; therefore, to consider this Function~~ opening ~~OPERABLE,~~ the turbine bypass valves ~~must remain shut at THERMAL POWER ≥ 30% RTP.~~ may affect the OPERABILITY of this Function 8 C

The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ ~~30~~ 45% RTP. This Function is not required when THERMAL POWER is < ~~30~~% RTP, since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Fixed Neutron Flux—High Functions are adequate to maintain the necessary safety margins. 3

INSERT Function 10

7 11 ~~10~~. Reactor Mode Switch—Shutdown Position

(A3 and B3)

The Reactor Mode Switch—Shutdown Position Function provides signals, via the two manual scram logic channels ~~to each of the four RPS logic channels,~~ which are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis. 3

The reactor mode switch is a single switch with ~~four~~ two channels, each of which provides input into one of the two manual scram RPS logic channels.

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

(continued)

7 INSERT Note 2

Note 2 has been provided to modify the ACTIONS for the RPS instrumentation functions of APRM Flow Biased Neutron Flux-High (Function 2.b) and APRM Fixed Neutron Flux — High (Function 2.c) when they are inoperable due to failure of SR 3.3.1.1.2 and gain adjustments are necessary. Note 2 allows entry into associated Conditions and Required Actions to be delayed for up to 2 hours if the APRM is indicating a lower power value than the calculated power (i.e., the gain adjustment factor (GAF) is high (non-conservative)), and for up to 12 hours if the APRM is indicating a higher power value than the calculated power (i.e., the GAF is low (conservative)). The GAF for any channel is defined as the power value determined by the heat balance divided by the APRM reading for that channel. Upon completion of the gain adjustment, or expiration of the allowed time, the channel must be returned to OPERABLE status or the applicable Condition entered and the Required Actions taken. This Note is based on the time required to perform gain adjustments on multiple channels and additional time is allowed when the GAF is out of limits but conservative. C

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

accurately reflects the required setpoint as a function of ~~flow.~~ Each flow signal from the respective flow ~~unit~~ converter must be ≤ ~~105%~~ 100 of the calibrated flow signal. If the flow ~~unit~~ converter signal is not within the limit, ~~one~~ all required APRM that receive(s) an input from the inoperable flow ~~unit~~ must be declared inoperable. 3

The Frequency of 7 days is based on engineering judgment, operating experience, and the reliability of this instrumentation.

SR 3.3.1.1.4 and SR 3.3.1.1.8 6

TSTF -205

INSERT SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. 2 C

As noted, SR 3.3.1.1.4 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM and APRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within ~~12~~ 24 hours after entering MODE 2 from MODE 1. ~~Twelve~~ twenty four hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR. 7

A Frequency of 7 days for SR 3.3.1.1.4 provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. ~~9~~ 13). 7 3

INSERT SR 3.3.1.1.8

7

~~SR 3.3.1.1.5~~ TSTF-205 not shown C

INSERT SR 3.3.1.1.5

~~A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A Frequency of 7 days provides an acceptable level of system average availability over the~~

(continued)

TSTF-205

INSERT SR 3.3.1.1.4

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

7

INSERT SR 3.3.1.1.8

The Frequency of 31 days for SR 3.3.1.1.8 is acceptable based on engineering judgment, operating experience, and the reliability of this instrumentation.

7

INSERT SR 3.3.1.1.5

A functional test of each automatic scram contactor is performed to ensure that each automatic RPS logic channel will perform the intended function. There are four RPS channel test switches, one associated with each of the four automatic trip channels (A1, A2, B1, and B2). These test switches allow the operator to test the OPERABILITY of the individual trip logic channel automatic scram contactors as an alternative to using an automatic scram function trip. This is accomplished by placing the RPS channel test switch in the test position, which will input a trip signal into the associated RPS logic channel. The RPS channel test switches are not specifically credited in the accident analysis. The Manual Scram Functions are not configured the same as the generic model used in Reference 13. However, Reference 13 concluded that the Surveillance Frequency extensions for RPS Functions were not affected by the difference in configuration since each automatic RPS logic channel has a test switch which is functionally the same as the manual scram switches in the generic model. As such, a functional test of each RPS automatic scram contactor using either its associated test switch or by test of any of the associated automatic RPS Functions is required to be performed once every 7 days. The Frequency of 7 days is based on the reliability analysis of Reference 13.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.6 and SR 3.3.1.1.7 (continued)

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.8 9 7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The ~~1000 MWD/T~~ Frequency is based on operating experience with LPRM sensitivity changes.

7 2000 effective full power hours (EFPH)

Insert move from Page B 3.3-30 7

SR 3.3.1.1.9 11 and SR 3.3.1.1.12 16 7

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.9 is based on the reliability analysis of Reference 9. 2 of SR 3.3.1.1.16

INSERT SR 3.3.1.1.11 7 TSTF -205 2 C

7 11 3 13 7 24

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. 24 7

SR 3.3.1.1.10 12 7

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be

(continued)

TSTF -205

<u>INSERT SR 3.3.1.1.11</u>

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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.3.1.1 - RPS INSTRUMENTATION

1. Typographical/grammatical error corrected.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

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

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

5. This Table has been deleted since it provides generic and not plant specific types of information. The information in the Table could be misleading as to which plant specific analyses take credit for these channels to perform a function during accident and transient scenarios.

6. Changes have been made to more closely reflect the Specification requirements.

7. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

8. The words have been modified to state that opening the bypass valves may affect the OPERABILITY of this Function. If the bypass valves are open above 45% RTP, but the Function is still enforcing the scram (i.e., it is not bypassed), there is no reason to declare the Function inoperable. If the Function is bypassed above 45% RTP due to an open bypass valve, then the Function would be inoperable. The proposed words state that an open bypass valve could affect the OPERABILITY of this Function. The words in the Bases for proposed SR 3.3.1.1.14 (ISTS SR 3.3.1.1.16) have been modified to state that the bypass valves must remain closed during the calibration if using actual turbine first stage pressure. At other times, the bypass valves can be open (and the bypass valves are periodically opened to perform SRs) as long as the Function is not inadvertently bypassed.

9. This Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

10. The bracketed item has been deleted.

BASES

APPLICABLE SAFETY ANALYSES (continued)

System (RPS) Instrumentation"; IRM Neutron Flux—High and Average Power Range Monitor (APRM) Neutron Flux—High, Setdown Functions; and LCO 3.3.2.1, "Control Rod Block Instrumentation."

1 U The SRMs have no safety function and are not assumed to function during any UFSAR design basis accident or transient analysis. However, the SRMs provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in Technical Specifications.

LCO

During startup in MODE 2, three of the four SRM channels are required to be OPERABLE to monitor the reactor flux level prior to and during control rod withdrawal, subcritical multiplication and reactor criticality, and neutron flux level and reactor period until the flux level is sufficient to maintain the IRM on Range 3 or above. All but one of the channels are required in order to provide a representation of the overall core response during those periods when reactivity changes are occurring throughout the core.

In MODES 3 and 4, with the reactor shut down, two SRM channels provide redundant monitoring of flux levels in the core.

In MODE 5, during a spiral offload or reload, an SRM outside the fueled region will no longer be required to be OPERABLE, since it is not capable of monitoring neutron flux in the fueled region of the core. Thus, CORE ALTERATIONS are allowed in a quadrant with no OPERABLE SRM in an adjacent quadrant provided the Table 3.3.1.2-1, footnote (b), requirement that the bundles being spiral reloaded or spiral offloaded are all in a single fueled region containing at least one OPERABLE SRM is met. Spiral reloading and offloading encompass reloading or offloading a cell on the edge of a continuous fueled region (the cell can be reloaded or offloaded in any sequence). C

In nonspiral routine operations, two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2.4

This Surveillance consists of a verification of the SRM instrument readout to ensure that the SRM reading is greater than a specified minimum count rate, which ensures that the detectors are indicating count rates indicative of neutron flux levels within the core. With few fuel assemblies loaded, the SRMs will not have a high enough count rate to satisfy the SR. Therefore, allowances are made for loading sufficient "source" material, in the form of irradiated fuel assemblies, to establish the minimum count rate.

with the detector full in 1

To accomplish this, the SR is modified by a Note that states that the count rate is not required to be met on an SRM that has less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With four or less fuel assemblies loaded around each SRM and no other fuel assemblies in the associated core quadrant, even with a control rod withdrawn, the configuration will not be critical.

1 When movable detectors are being used, detector location must be selected such that each group of fuel assemblies is separated by at least two fuel cells from any other fuel assemblies.

The Frequency is based upon channel redundancy and other information available in the control room, and ensures that the required channels are frequently monitored while core reactivity changes are occurring. When no reactivity changes are in progress, the Frequency is relaxed from 12 hours to 24 hours.

TSTF-205

SR 3.3.1.2.5 and SR 3.3.1.2.6

C

INSERT SR 3.3.1.2.5

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the ~~associated channel will function properly.~~ SR 3.3.1.2.5 is required in MODE 5, and the 7 day Frequency ensures that the channels are OPERABLE while core reactivity changes could be in progress. This Frequency is reasonable, based on operating experience and on other Surveillances (such as a CHANNEL CHECK), that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

to be met 5

SR 3.3.1.2.6 is required in MODE 2 with IRMs on Range 2 or below, and in MODES 3 and 4. Since core reactivity changes ~~do not normally take place~~, the Frequency ~~has been~~ is extended from 7 days to 31 days. The 31 day Frequency is based on operating experience and on other Surveillances (such as

1 in MODES 3 and 4 and core reactivity changes are due only to control rod movement in MODE 2

(continued)

TSTF -205

<u>INSERT SR 3.3.1.2.5</u>

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

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~~ 11) 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

1

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~~ 12).

1C

1

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

1C

(continued)

TSTF -205

INSERT SR

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

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.

2 on a startup and entry into MODE 2 concurrent with a reduction to ≤10% RTP during a shutdown

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 ~~.~~ 1 and by attempting to select a control rod not in compliance with the prescribed sequence and verifying a selection error 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 2 and 4 at ≤10% RTP in MODE 2. ~~As noted,~~ 2 The Note to SR 3.3.2.1.2 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).~~ Insert SR 3.3.2.1.2 1

6 Insert from pages B 3.3-53 and B 3.3-54

SR 3.3.2.1.~~4~~ 5 6

Insert SR 3.3.2.1.5 1

~~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~~ bypass setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the ~~power range~~ APRM channel can be placed in the conservative condition ~~(i.e., enabling the proper RBM setpoint)~~ to enable the RBM. 1 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~~ 9 6. The ~~18~~ 24 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.5 6 6 (continued)

3 7

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 3.3.2.1.6 7 6

TSTF -205

INSERT 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. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch—Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

1C

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 18 24 month 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.

6

The 18 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 24 month Frequency.

6

move to page B 3.3-52 as indicated

SR 3.3.2.1.7 4 6

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel

(continued)

TSTF -205

INSERT SR 3.3.2.1.7

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

6 Insert SR 3.3.2.1.9

SR 3.3.2.1.9

LCO 3.1.3 and LCO 3.1.6 may require individual control rods to be bypassed (taken out of service) in the RWM to allow insertion of an inoperable control rod or correction of a control rod pattern not in compliance with the analyzed rod position sequence. With the control rods bypassed (taken out of service) in the RWM, the RWM will provide insert and withdraw blocks for bypassed control rods that are fully inserted and a withdraw block for bypassed control rods that are not fully inserted. To ensure the proper bypassing and movement of these affected control rods, 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) must verify the bypassing and position of these control rods. Compliance with this SR allows the RWM to be OPERABLE with these control rods bypassed. C C

1 Insert REF

4. UFSAR, Section 15.4.10.

5. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1 Exxon Nuclear Methodology for Boiling Water Reactor-Neutronics Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).

6. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (as specified in Technical Specification 5.6.5).

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

8. NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).

BASES

ACTIONS (continued)

C.1 and C.2

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 25% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below 25% RTP results in sufficient margin to the required limits, and the feedwater System and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. Alternatively, if a channel is inoperable solely due to an inoperable feedwater pump breaker, the affected feedwater pump breaker may be removed from service since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 25% RTP from full power conditions in an orderly manner and without challenging plant systems. 1 2

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 Safety Evaluation Report (SER) for the topical report.~~ 5

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater System and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumption that 6 hours is the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the feedwater pumps ~~turbines~~ and main turbine will trip when necessary. 1 2 3 C

SR 3.3.2.2.1

Performance of the CHANNEL CHECK once every ~~24~~ 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter 1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.2.1 (continued)

indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels, or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

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

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

TSTF -205

SR 3.3.2.2.2

INSERT SR3.3.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. 2 C

The Frequency of 92 days is based on reliability analysis (Ref. 2).

1 Insert SR 3.3.2.2.3

SR 3.3.2.2.3 4

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

(continued)

TSTF -205

INSERT SR 3.3.2.2.2

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

3

Insert SR 3.3.2.2.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in SR 3.3.2.2.4. If the trip setting is discovered to be less conservative than the setting accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on engineering judgement and the reliability of these components.

All changes are [1] unless noted otherwise

BASES

LCO

2. Reactor Vessel Water Level (continued)

reactor vessel

a specific vessel

The ~~wide range~~ water level instruments are uncompensated for variation in reactor water density and are calibrated to be most accurate at ~~operational~~ pressure and temperature.

The wide range instruments are calibrated to be accurate at post-DBA LOCA pressure and temperature. The medium range instruments are calibrated to be accurate at the normal operating pressure and temperature.

Torus

3. ~~Suppression Pool~~ Water Level

Type A and

~~Suppression pool~~ water level is a Category I variable provided to detect a breach in the reactor coolant pressure boundary (RCPB). This variable is also used to verify and provide long term surveillance of ECCS function. The wide range ~~suppression pool~~ water level measurement provides the operator with sufficient information to assess the status of both the RCPB and the water supply to the ECCS. The wide range water level indicators monitor the ~~suppression pool~~ water level from the ~~center line of the ECCS suction lines~~ to the top of the ~~pool~~. Two wide range ~~suppression pool~~ water level signals are transmitted from separate differential pressure transmitters and ~~are~~ continuously ~~recorded~~ on two recorders in the control room. These ~~recorders~~ are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

torus

bottom

to two control room indicators

also

displayed

instruments

4. Drywell Pressure

Type A and

The wide range drywell pressure

Drywell pressure is a Category I variable provided to detect breach of the RCPB and to verify ECCS functions that operate to maintain RCS integrity. Two ~~wide~~ range ~~drywell pressure~~ signals are transmitted from separate pressure transmitters and are continuously recorded ~~and displayed~~ on two control room recorders. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

a

different

s

channels provide the PAM Drywell Pressure Function.

The wide range instruments measure from -5 psig to 250 psig while the narrow range instruments monitor between -5 psig and 70 psig.

and displayed on two control room indicators

and indicators

Insert LCO4

5. ~~Primary Containment Area~~ Radiation ~~(High Range)~~

a Category I variable

~~Primary containment area~~ radiation ~~(High range)~~ is provided to monitor the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. ~~[For~~

Drywell

(continued)

The drywell pressure channels also satisfy the Reference 2 monitoring requirement for suppression chamber (torus) pressure (a Type A and Category I variable) since the suppression chamber-to-drywell vacuum breakers ensure the suppression chamber pressure is maintained within 0.5 psig of drywell pressure.

C

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary.

SR 3.3.4.1.1

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

TSTF-205 Not shown C

The Frequency of 92 days is based on reliability analysis of Reference 5.

SR 3.3.4.1.2

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in SR 3.3.4.1.3. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on assumptions of the reliability analysis (Ref. 5) and on the methodology included in the determination of the trip setpoint.

SR 3.3.4.1.3

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.2.1 (continued)

something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

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

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

INSERT SR 3.3.4.1.3 TSTF-205

SR 3.3.4.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 2.

move to page B 3.3-99 as indicated

SR 3.3.4.2.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in SR 3.3.4.2.4. If the trip setting is discovered to be less conservative than the setting accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the

(continued)

TSTF -205

INSERT SR 3.3.4.1.3

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.

Insert BKGD-3

The LPCI System initiation logic also contains LPCI Loop Select Logic whose purpose is to identify and direct LPCI flow to the unbroken recirculation loop if a Design Basis Accident (DBA) occurs. The LPCI Loop Select Logic is initiated upon the receipt of either a LPCI Reactor Vessel Water Level - Low Low signal or a LPCI Drywell Pressure - High signal, as discussed previously. When initiated, the LPCI Loop Select Logic first determines recirculation pump operation by sensing the differential pressure (dp) between the suction and discharge of each pump. There are four dp switches monitoring each recirculation loop which are, in turn, connected to relays whose contacts are connected to two trip systems. The dp switches will trip when the dp across the pump is approximately 2 psid. The contacts are arranged in a one-out-of-two taken twice logic for each recirculation pump. If the logic senses that either pump is not running, i.e., single loop operation, then a trip signal is sent to both recirculation pumps to eliminate the possibility of pipe breaks being masked by the operating recirculation pump pressure. However, the pump trip signal is delayed approximately 0.5 seconds (one time delay relay per trip system) to ensure that at least one pump is off since the break detection sensitivity is greater with both pumps running. If a pump trip signal is generated, reactor steam dome pressure must drop to a specified value before the logic will continue. This adjusts the selection time to optimize sensitivity and still ensure that LPCI injection is not unnecessarily delayed. The reactor steam dome pressure is sensed by four pressure switches which in turn are connected to relays whose contacts are connected to two trip systems. The contacts are arranged in a one-out-of-two taken twice logic. After the satisfaction of this pressure requirement or if both recirculation pumps indicate they are running, a 2 second time delay is provided to allow momentum effects to establish the maximum differential pressure for loop selection. Selection of the unbroken recirculation loop is then initiated. This is done by comparing the absolute pressure of the two recirculation riser loops. The broken loop is indicated by a lower pressure than the unbroken loop. The loop with the higher pressure is then used for LPCI injection. If, after a small time delay of approximately 0.5 seconds (one time delay relay per trip system), the pressure in loop A is not indicating higher than loop B, the logic will provide a signal to close the B recirculation loop discharge valve, open the LPCI injection valve to the B recirculation loop and close the LPCI injection valve to the A recirculation loop. This is the "default" choice in the LPCI Loop Select Logic. If recirculation loop A pressure indicates higher than loop B pressure, approximately 2 psig, the recirculation discharge valve in loop A is closed, the LPCI injection valve to loop A is signaled to open and the LPCI injection valve to loop B is signaled to close. The four dp switches which provide input to this portion of the logic detect the pressure difference between the corresponding risers to the jet pumps in each recirculation loop. The four dp switches are connected to relays whose contacts are connected to two trip systems. The contacts in each trip system are arranged in a one-out-of-two taken twice logic. There are two redundant trip systems in the LPCI Loop Select Logic. The complete logic in each trip system must actuate for operation of the LPCI Loop Select Logic.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.1

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

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

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

TSTF -205

INSERT SR3.3.5.1.2

SR 3.3.5.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. 1

1

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of Reference 5.

(continued)

TSTF -205

INSERT SR 3.3.5.1.2

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

C

all changes are [1] unless otherwise identified

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.3.5.2.1 (continued)~~

~~something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.~~

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

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

SR 3.3.5.2.~~2~~ 1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. [4] INSERT SR 3.3.5.2.1 [TSTF -205] [C]

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of ~~92~~ 31 days is based on ~~the reliability analysis of Reference 1~~.

[2] Plant operating experience with regard to channel OPERABILITY and drift that demonstrates that failure of more than one channel in any 31 day interval is rare

~~SR 3.3.5.2.3~~

~~The calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.5.2-1. If the trip setting is discovered to be less conservative than the setting accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint~~

(continued)

TSTF -205

INSERT SR 3.3.5.2.1

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

I INSERT BKGD-3

The Drywell Radiation - High Function receives input from two radiation detector assemblies each connected to a switch. Each switch actuates two contacts. Each contact inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the PCIVs identified in Reference 1. The two contacts associated with the same switch provide input to both trip strings in the same trip system. Any contact will trip the associated trip string. The trip strings are arranged in a one-out-of-two taken twice logic. For the purpose of this Specification, a channel is considered to include a radiation detector assembly, a switch, and one of two contacts.

I INSERT BKGD-4

The Isolation Condenser Steam Flow—High and Return Flow—High Functions each receive input from one channel with its associated flow switch. The steam flow switch and the condensate flow switch are connected in a one-out-of-two logic in each of two trip strings. Each of the two trip strings provides input into two trip systems in a one-out-of-two logic and each trip system isolates either the inboard or outboard Isolation Condenser steam and condensate isolation valves. For the purpose of this Specification, an Isolation Condenser Steam Flow—High Function channel and the associated Return Flow—High channel must be OPERABLE (one separate channel for each trip system).

I INSERT BKGD-5

The HPCI Turbine Area Temperature—High Function receives input from 16 temperature switches. Four channels, each with an associated temperature switch, provide inputs to a one-out-of-two-twice logic arrangement in each of two AC and two DC trip strings. Each of the trip strings provides input into both an AC and DC trip system and only one trip string must trip to trip the associated trip system. However for OPERABILITY, only one DC trip string is required to provide input into the DC trip system and only one AC trip string is required to provide input into the AC trip system. Either trip system isolates both valves in the HPCI steam supply penetration. C

1 Insert BKGD-6

Each channel inputs into one of four trip stings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the reactor water cleanup (RWCU) valves. Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation.

1 Insert BKGD-7

the SLC initiation switch. The switch provides trip signal inputs to both trip systems in any position other than "OFF". The other switch positions are SYS 1, SYS 2, SYS 1+2 and SYS 2+1. For the purpose of this Specification, the SLC initiation switch is considered to provide 1 channel input into each trip system.

1 Insert BKGD-8

Each channel inputs into one of four trip stings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the SDC suction isolation valves. Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation.

4 Insert Function 1.c

1.c. Main Steam Line Pressure - Timer

Main Steam Line Pressure—Timer is provided to prevent false isolations on low MSL pressure as a result of pressure transients, however, the timer must function in a limited time period to support the OPERABILITY of the Main Steam Line Pressure—Low Function by enabling the associated channels after a certain time delay. The Main Steam Line Pressure—Timer is directly assumed in the analysis of the pressure regulator failure (Ref. 5). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. 1 C

The MSL low pressure timer signals are initiated when the associated MSL low pressure switch actuates. Four channels of Main Steam Line Pressure—Timer Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to be long enough to prevent false isolations due to pressure transients but short enough to prevent excessive RPV depressurization.

This Function isolates the Group 1 valves.

All changes are 1 unless noted otherwise

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

.1

3.~~e., 3.f., 3.h., 3.i., 4.e., 4.g., 4.h., 4.i., 4.j.~~ d HPCI Turbine Area ~~and Differential~~ Temperature—High (continued) 4

~~Area and Differential~~ HPCI Turbine Area Temperature—High signals are initiated from ~~thermocouples~~ temperature switches that are appropriately located to ~~protect the system~~ detect a leak from the system piping that is being monitored. ~~Two instruments monitor each area. Two channels for each HPCI and RCIC Area and Differential Temperature—High Function are available and are required to be OPERABLE~~ Sixteen instruments monitor the HPCI Turbine Area. Sixteen channels for HPCI Turbine Area Temperature - High are available, however only eight channels are required to be OPERABLE 4 Four As Noted (footnote (a) to Table 3.3.6.1-1) each trip system associated with this Function requires all four channels to be associated with a single trip string. (four channels within the same AC trip string for the AC trip system and four channels within the same DC trip string for the DC trip system) C to ensure that no single instrument failure can preclude the isolation function.

~~Eight thermocouples provide input to the Area Ventilation Differential Temperature—High Function. The output of these thermocouples is used to determine the differential temperature. Each channel consists of a differential temperature instrument that receives inputs from thermocouples that are located in the inlet and outlet of the area cooling system for a total of four available channels (two for RCIC and two for HPCI).~~ 1 4

The Allowable Value~~s are~~ is set well above the expected ambient condition but low enough to detect ~~a leak equivalent to 25 gpm~~ steam line leakage. C

These Functions isolate the Group ~~3 and~~ 4 valves~~, as appropriate~~.

~~3.g., 4.f. Suppression Pool Area Temperature—Time Delay Relay~~ 1 4

~~The Suppression Pool Area Temperature—Time Delay Relays are provided to allow all the other systems that may be leaking into the pool area (as indicated by the high temperature) to be isolated before HPCI and/or RCIC are automatically isolated. This ensures maximum HPCI and RCIC System operation by preventing isolations due to leaks in other systems. These Functions are not assumed in any FSAR transient or accident analysis.~~

~~There are four time delay relays (two for HPCI and two for RCIC). Two channels each for both HPCI and RCIC Suppression Pool Area Temperature—Time Delay Relay Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.1 (continued)

CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

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

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

SR 3.3.6.1.2 and SR 3.3.6.1.5

TSTF -205

INSERT SR 3.3.6.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analysis described in References 5 and 7. The ~~184 day~~ 24 month Frequency of SR 3.3.6.1.5 is based on engineering judgment and the reliability of the components ~~(time delay relays exhibit minimal drift)~~.

e 6 8 9 1 2 4

(continued)

TSTF-205

INSERT SR 3.3.6.1.2

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.6.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than that accounted for in the appropriate setpoint methodology.

6 — e — The Frequency of 92 days ~~is~~ (8) is based on the reliability analys~~is~~ of References (8) and (9) — 1

1c

SR 3.3.6.1.4 and SR 3.3.6.1.6

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

24 — 4 — The Frequency of SR 3.3.6.1.4 is based on the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.6.1.6 is based on the assumption of ~~an~~ a ~~18~~ 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. — 4 1c

1c

SR 3.3.6.1.7

24 — 4 — The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The ~~18~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the

(continued)

BASES

Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.7 (continued) 6

Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 24 month Frequency. 4

1C

~~SR 3.3.6.1.8~~

~~This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. Testing is performed only on channels where the assumed response time does not correspond to the diesel generator (DG) start time. For channels assumed to respond within the DG start time, sufficient margin exists in the [10] second start time when compared to the typical channel response time (milliseconds) so as to assure adequate response without a specific measurement test. The instrument response times must be added to the PCIV closure times to obtain the ISOLATION SYSTEM RESPONSE TIME.~~

~~ISOLATION SYSTEM RESPONSE TIME acceptance criteria are included in Reference 7. This test may be performed in one measurement, or in overlapping segments, with verification that all components are tested.~~

~~A Note to the Surveillance states that the radiation detectors may be excluded from ISOLATION SYSTEM RESPONSE TIME testing. This Note is necessary because of the difficulty of generating an appropriate detector input signal and because the principles of detector operation virtually ensure an instantaneous response time. Response times for radiation detector channels shall be measured from detector output or the input of the first electronic component in the channel.~~

~~ISOLATION SYSTEM RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. The 18 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience that shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2.1 (continued)

channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

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

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

TSTF-205

SR 3.3.6.2.2

INSERT SR 3.3.6.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. C

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. 2

The Frequency of 92 days is based on the reliability analysis of References 5 and 6. 4 5 1

SR 3.3.6.2.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.6.2-1. If the trip setting is discovered to be less conservative than

(continued)

TSTF -205

INSERT SR 3.3.6.2.2

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

C

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.3.6.3.2, SR 3.3.6.3.3, and SR 3.3.6.3.4~~

TSTF-205 not shown

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

~~The 92 day Frequency is based on the reliability analysis of Reference 3.~~

~~A portion of the S/RV tailpipe pressure switch instrument channels are located inside the primary containment. The Note for SR 3.3.6.3.3, "Only required to be performed prior to entering MODE 2 during each scheduled outage > 72 hours when entry is made into primary containment," is based on the location of these instruments, ALARA considerations, and compatibility with the Completion Time of the associated Required Action (Required Action B.1).~~

C

~~SR 3.3.6.3.5~~

~~The calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than the setting accounted for in the appropriate setpoint methodology. The Frequency of every 92 days for SR 3.3.6.3.5 is based on the reliability analysis of Reference 3.~~

1

SR 3.3.6.3.6 1 and SR 3.3.6.3.2

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

(continued)

All changes are $\boxed{1}$ unless otherwise indicated

CREV

B 3.3 INSTRUMENTATION

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

Emergency Ventilation (CREV)

BASES

CREV

BACKGROUND

The ~~MCREC~~ System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. ~~Two independent MCREC subsystems are each~~ capable of fulfilling the stated safety function. The instrumentation ~~and controls for~~ the ~~MCREC~~ System ~~automatically initiate action to~~ pressurize the ~~main~~ control room ~~(MCR)~~ to minimize the consequences of radioactive material in the control room environment.

The CREV System is

CREV System

CREV

and

provides control room alarms so that manual action can be taken to start

emergency zone

In the event of a ~~loss of coolant accident (LOCA) signal (Reactor Vessel Water Level—Low Low Low, Level 1 or Drywell Pressure—High), Main Steam Line Flow—High, Refueling Floor Area Radiation—High, or Control Room Air Inlet Radiation—High~~ signal, ~~the MCREC System is automatically started in the pressurization mode.~~ ~~The air is then recirculated through the charcoal filter, and sufficient outside air is drawn in through the normal intake to maintain the MCR slightly pressurized with respect to the turbine building.~~

Reactor Building Ventilation System

High Radiation alarm

adjacent zones

control room emergency zone

A description of the CREV System is provided in the Bases for LCO 3.7.4, "Control Room Emergency Ventilation (CREV) system."

The ~~MCREC~~ System instrumentation has two trip systems, either ~~of which can initiate both MCREC subsystems (Ref. 1)~~. Each trip system receives input from ~~each of the functions listed above.~~ ~~The Functions are arranged as follows for each trip system. The Reactor Vessel Water Level—Low Low Low, Level 1 and Drywell Pressure—High are each arranged in a one-out-of-two taken twice logic (these signals are the same that start the low pressure Emergency Core Cooling Systems' (ECCS) subsystems). The Main Steam Line Flow—High is arranged in a one-out-of-four taken twice logic (each main steam line has two high flow inputs to the trip system). The Refueling Floor Area Radiation—High and Control Room Air Inlet Radiation—High are each arranged in a one-out-of-one logic.~~ The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a ~~MCREC System initiation~~ signal to the ~~initiation~~ logic.

CREV

of which provide sufficient information to ensure the CREV System is initiated and the dampers are closed when necessary

Insert BKGD

alarm

(continued)

operator action is required to switch the CREV System to the isolation/pressurization mode of operation and close required dampers

1 Insert BKGD

one radiation monitor channel. Two detectors (one detector for each radiation monitor channel) are located in the reactor building ventilation exhaust duct. The output of each channel is provided to one trip system (i.e., one radiation monitor channel per trip system). The output from each channel is arranged in a one-out-of-one trip (alarm) system. A trip of any trip system will initiate a Reactor Building Ventilation System - High High Radiation Alarm in the control room.

1 Insert LCO-1

High reactor building ventilation exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When high reactor building ventilation exhaust radiation is alarmed in the control room, the CREV System is manually initiated in the isolation/pressurization mode and required dampers are closed since this condition could result in radiation exposure to control room personnel.

The Reactor Building Ventilation System - High High Radiation Alarm Function signals are initiated from radiation detectors that are located in the ventilation exhaust ducting coming from the reactor building and refueling zones. The signals from each detector are input to individual monitors whose trip outputs are assigned to a control room alarm. Two channels of Reactor Building Ventilation System - High High Radiation Alarm Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the alarm function. The Allowable Value was selected to promptly detect gross failure of the fuel cladding and to ensure protection of control room personnel. C

1 Insert LCO-2

The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

CREV 1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1.1 (continued)

outside the criteria, it may be an indication that the instrument has drifted outside its limit.

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

SR 3.3.7.1.2

TSTF -205

INSERT SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. 2 1C

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of References ~~5~~ 3 and ~~6~~ 4.

~~SR 3.3.7.1.3~~

~~The calibration of trip units provides a check of the actual trip setpoints. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.7.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than the setting accounted for in the appropriate setpoint methodology.~~ 3

~~The Frequency of 92 days is based on the reliability analyses of References 5 and 6.~~

(continued)

TSTF -205

INSERT SR 3.3.7.1.2

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

~~SR 3.3.8.1.1~~

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

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

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

4

TSTF -205

INSERT SR 3.3.8.1.1

4 1 and SR 3.3.8.1.3 4

SR 3.3.8.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. 1

18 months and 24 months are 4

ies 4

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

18 month or 24 month

4 , as applicable,

B

C

B

B

(continued)

TSTF -205

INSERT SR 3.3.8.1.1

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.

BASES

BACKGROUND (continued)

circuit breakers has an associated independent set of Class 1E overvoltage, undervoltage, and underfrequency sensing logic. Together, a circuit breaker and its sensing logic constitute an electric power monitoring assembly. If the output of the inservice MG set or alternate power supply exceeds predetermined limits of overvoltage, undervoltage, or underfrequency, a trip coil (undervoltage release coil) within the circuit breaker driven by this logic circuitry opens the circuit breaker, which removes the associated power supply from service.

3 inservice

1

APPLICABLE SAFETY ANALYSES

3

The RPS ~~e~~lectric ~~p~~ower ~~m~~onitoring is necessary to meet the assumptions of the safety analyses by ensuring that the RPS equipment powered from the RPS buses can perform its intended function. RPS ~~e~~lectric ~~p~~ower ~~m~~onitoring provides protection to the RPS ~~and other systems that receive power from the RPS buses~~ bus components, by acting to disconnect the RPS from the power supply under specified conditions that could damage the RPS ~~bus powered~~ equipment.

1 RPS components

1

RPS ~~e~~lectric ~~p~~ower ~~m~~onitoring satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

1

LCO

The OPERABILITY of each RPS electric power monitoring assembly is dependent on the OPERABILITY of the overvoltage, undervoltage, and underfrequency logic, as well as the OPERABILITY of the associated circuit breaker. Two electric power monitoring assemblies are required to be OPERABLE for each inservice power supply. This provides redundant protection against any abnormal voltage or frequency conditions to ensure that no single RPS electric power monitoring assembly failure can preclude the function of RPS bus powered components. Each of the inservice electric power monitoring assembly ~~'s~~ trip logic setpoints ~~are~~ is required to be within the specified Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

3

Allowable Values are specified for each RPS electric power monitoring assembly trip logic (refer to SR 3.3.8.2.2). 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

(continued)

BASES

ACTIONS

D.1, ~~D.2.1, and D.2.2~~ (continued) 4

~~In addition, action must be immediately initiated to either restore one electric power monitoring assembly to OPERABLE status for the inservice power source supplying the required instrumentation powered from the RPS bus (Required Action D.2.1) or to isolate the RHR Shutdown Cooling System (Required Action D.2.2). Required Action D.2.1 is provided because the RHR Shutdown Cooling System may be needed to provide core cooling. All actions must continue until the applicable Required Actions are completed.~~

TSTF -205

SURVEILLANCE REQUIREMENTS

INSERT SR 3.3.8.2.1

3

SR 3.3.8.2.1

A CHANNEL FUNCTIONAL TEST is performed on each overvoltage, undervoltage, and underfrequency channel to ensure that the ~~entire~~ channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted in the Surveillance, the CHANNEL FUNCTIONAL TEST is only required to be performed while the plant is in a condition in which the loss of the RPS bus will not jeopardize steady state power operation (the design of the system is such that the power source must be removed from service to conduct the Surveillance). The 24 hours is intended to indicate an outage of sufficient duration to allow for scheduling and proper performance of the Surveillance.

The 184 day Frequency and the Note in the Surveillance are based on guidance provided in Generic Letter 91-09 (Ref. 2).

SR 3.3.8.2.2

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

(continued)

TSTF
-205

INSERT 3.3.8.2.1

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

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?

   This change removes the requirement to notify the NRC if required by 10 CFR 50.72 and to submit a Licensee Event Report as required by 10 CFR 50.73 if the inoperable channels are not replaced in the trip condition in MODES 4 and 5. The change replaces these requirements with a specific action to declare the associated subsystems (i.e., the associated ECCS subsystems) inoperable, thus requiring the actions for an inoperable ECCS subsystem to be taken. The inoperable ECCS subsystem actions have been previously approved by the NRC (as modified by the Discussion of Changes in this submittal). The required reports are not considered as initiators for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The consequences of a previously analyzed accident are not affected by the deletion of these reporting requirements since they do not impact the assumptions of any design basis accident or transient.

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?

   The margin of safety is not reduced by removing the requirement for the submittal of these required reports. This change has no effect on the assumptions of design basis accidents or transients. This change has no impact on safe operation of the plant because adequate actions are provided if the inoperable channels are not placed in the trip condition in MODES 4 and 5. This change does not affect any plant equipment or requirements for maintaining plant equipment. Therefore, this change does not involve a significant reduction in a margin of safety.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 Shutdown Cooling (SDC) System—Hot Shutdown

LCO 3.4.7 Two SDC subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one SDC subsystem shall be in operation.

-----NOTES-----

1. Both required SDC subsystems and recirculation pumps may be not in operation for up to 2 hours per 8 hour period.

2. One required SDC subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

APPLICABILITY: MODE 3, with reactor vessel coolant temperature less than the SDC cut-in permissive temperature.

ACTIONS

-----NOTES-----

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each SDC subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required SDC subsystems inoperable.	A.1 Initiate action to restore required SDC subsystem(s) to OPERABLE status. AND	Immediately (continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Shutdown Cooling (SDC) System—Cold Shutdown

LCO 3.4.8 Two SDC subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one SDC subsystem shall be in operation.

-----NOTES-----

1. Both required SDC subsystems may be not in operation during hydrostatic testing.
2. Both required SDC subsystems and recirculation pumps may be not in operation for up to 2 hours per 8 hour period.
3. One required SDC subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

APPLICABILITY: MODE 4.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each shutdown cooling subsystem.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two required SDC subsystems inoperable.	A.1	Verify an alternate method of decay heat removal is available for each inoperable required SDC subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No required SDC subsystem in operation. AND No recirculation pump in operation.	B.1 Verify reactor coolant circulating by an alternate method. AND B.2 Monitor reactor coolant temperature and pressure.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one SDC subsystem or recirculation pump is operating.	12 hours

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

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 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 3.4.5.2

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 3.4.5.2 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

LCO (continued)	common to both subsystems. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 3, one SDC subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. To ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. Note 1 permits both required SDC subsystems and recirculation pumps to not be in operation for a period of 2 hours in an 8 hour period. Note 2 allows one required SDC subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected SDC System or on some other plant system or component that necessitates placing the SDC System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the SDC subsystems or other operations requiring SDC flow interruption and loss of redundancy.
APPLICABILITY	In MODE 3 with reactor vessel coolant temperature below the SDC cut-in permissive temperature (i.e., the actual temperature at which the interlock resets) the SDC System may be operated in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature. Otherwise, a recirculation pump is required to be in operation. In MODES 1 and 2, and in MODE 3 with reactor vessel coolant temperature greater than or equal to the SDC cut-in permissive temperature, this LCO is not applicable. Operation of the SDC System in the shutdown cooling mode is not allowed above this temperature because the RCS temperature may exceed the design temperature of the 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.

(continued)

BASES

ACTIONS

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

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.

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 Systems 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).

However, due to the potentially reduced reliability of the alternate methods of decay heat removal, it is also required to reduce the reactor coolant temperature to the point where MODE 4 is entered.

B.1, B.2, and B.3

With no required SDC subsystem and no recirculation pump in operation, except as permitted by LCO Note 1, reactor coolant circulation by the SDC subsystem or recirculation pump must be restored without delay. 1C

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.

(continued)

BASES

LCO (continued)

component that is assumed not to fail, it is allowed to be common to both subsystems. Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one SDC subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. To ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

Note 1 allows both required SDC subsystems to not be in operation during hydrostatic testing. This allowance is acceptable because adequate reactor coolant circulation will be maintained by operation of a reactor recirculation pump to ensure adequate core flow and since systems are available to control reactor coolant temperature. Note 2 permits both SDC subsystems and recirculation pumps to not be in operation for a period of 2 hours in an 8 hour period. Note 3 allows one required SDC subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected SDC System or on some other plant system or component that necessitates placing the SDC System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the SDC subsystems or other operations requiring SDC flow interruption and loss of redundancy.

APPLICABILITY

In MODE 4, the SDC System must be OPERABLE and one SDC subsystem shall be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 212°F. Otherwise, a recirculation pump is required to be in operation.

In MODES 1 and 2, and in MODE 3 with reactor vessel coolant temperature greater than or equal to the SDC cut-in permissive temperature, this LCO is not applicable. Operation of the SDC System in the shutdown cooling mode is not allowed above this temperature because the RCS temperature may exceed the design temperature of the

(continued)

BASES

ACTIONS

A.1 (continued)

verification of the 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.

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

C. Recirculation Pumps

LCO 3.4.1 Recirculation pump speed shall be maintained within:

SR3.4.1.1 — L.4

1. 10% of each other with THERMAL POWER ≥80% of RATED THERMAL POWER.

2. 15% of each other with THERMAL POWER <80% of RATED THERMAL POWER.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2 ~~during two recirculation loop operation~~. A.2

ACTION:

Action B With the recirculation pump speeds different by more than the specified limits, ~~either~~:

1. ~~Restore the recirculation pump speeds to within the specified limit~~ within 2 hours, ~~or~~ A.5 L.2

2. ~~Trip one of the recirculation pumps and take the ACTION required by Specification 3.6.A.1.~~ A.6

4.6 - SURVEILLANCE REQUIREMENTS

C. Recirculation Pumps

SR 3.4.1.1 Recirculation pump speed shall be verified to be within the limits at least once per 24 hours.

add proposed SR 3.4.1.1 Note L.3

ADMINISTRATIVE

A.4 (cont'd) referenced Specifications 2.2.A and 3.2.E (ITS 3.3.1.1 and ITS 3.3.2.1) and replaced with Allowable Values, therefore, this change is considered administrative. The elimination of Trip Setpoints, and replacement with Allowable Values, will be addressed in the Discussion of Changes for ITS 3.3.1.1 and ITS 3.3.2.1.

A.5 CTS 3.6.C Action 1 requires restoration of the recirculation pump speeds (i.e., jet pump loop flow in ITS) to within the limits if they are not within the limits. The revised presentation of ITS ACTIONS (based on the BWR ISTS, NUREG-1433, Rev. 1) does not explicitly detail options to "restore...to within the specified limit" when an alternate ACTION is provided that allows continued operation. This action is always an option, and is implied in all ACTIONS. Since CTS 3.6.C Action 1 (ITS 3.4.1 ACTION B) provides an alternate action that allows continued operation, deleting CTS 3.6.C Action 1 is purely editorial.

A.6 CTS 3.6.C Action 2 requires action to be taken per CTS 3.6.A.1 when recirculation pump speeds differ by more than the specified limits. The format of the ITS does not include providing "cross references." CTS 3.6.A.1 (ITS 3.4.1) adequately prescribes the necessary conditions for compliance without such references. Therefore, the existing reference to "take the ACTION required by Specification 3.6.A.1" in CTS 3.6.C Action 2 serves no functional purpose, and its removal is purely an administrative difference in presentation.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 With no reactor coolant system recirculation loops in operation, CTS 3.6.A Action 2 requires the unit to be in at least STARTUP (MODE 2) within 8 hours and in HOT SHUTDOWN (MODE 3) within the next 6 hours. Under the same conditions ITS 3.4.1 Required Action A.1 will require the unit to be in MODE 2 in 8 hours and Required Action A.2 will require the unit to be in MODE 3 in 12 hours (next 4 hours). The change has been made for consistency with other conditions in the CTS and ITS which require the units to be in MODE 3. This change is more restrictive since the total time required to be in MODE 3 has decreased from 14 to 12 hours. This proposed time period is still adequate to achieve the required plant conditions in an orderly manner and without challenging plant systems.

C

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The detail of the actual MCPR correction factor for the MCPR operating limit for single loop operation ("0.01") in CTS 3.6.A Action 1.b is proposed to be relocated to the COLR. The requirement in proposed LCO 3.4.1 to apply the LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR during operation with one recirculation loop and the requirement in proposed ITS 3.4.1 ACTION C to satisfy the requirements of the LCO within 24 hours are adequate to ensure the current requirement is performed during single loop operation. Since all the requirements of CTS 3.6.A Action 1.b (except for the actual limit) are maintained in the proposed specification, the proposed changes are considered adequate. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the COLR will be controlled by the provisions of the COLR change control process described in Chapter 5 of the ITS.

LA.2 CTS 4.6.A requires the recirculation pump MG set scoop tube stop settings specified in the COLR to be verified at least once per 18 months. As indicated in the CTS requirement, the scoop tube stop settings are currently specified in the COLR. The details related to these operational settings are proposed to be relocated to Technical Requirements Manual (TRM). The MCPR operating limit is dependent on the MG set scoop tube stop settings as indicated in the Bases of ITS 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR). Therefore, with the MG set scoop tube stop settings not within limit, the MCPR operating limit may not be valid and therefore MCPR must be declared not within limits in accordance with proposed ITS 3.2.2 Required Action A.1 and action must be taken to restore MCPR to within limits within 2 hours or the THERMAL POWER must be reduced below 25% RTP (ITS 3.2.2 Required Action B.1). SR 3.2.2.1 requires the MCPRs to be verified to be greater than the limits specified in the COLR once within 12 hours after THERMAL POWER is $\geq$ 25% RTP and once per 24 hours thereafter. The MCPR limits specified in the COLR are based on MG set scoop tube settings. Therefore, if the MG set scoop tube settings are not set in accordance with the relocated requirement, the MCPR must be declared not within limits. These controls are considered adequate to ensure that MCPR will be within limit during normal and transient conditions. During transients initiated at reduced core flow the transient analysis assumes a failed speed rate (not speed limit) controller which results in an infinitely slow recirculation pump run-up rate which results in the most limiting MCPR. Most failures in the recirculation flow control system would actually

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 (cont'd) result in a faster transient which will be mitigated by the Average Power Range Monitor Flow Biased Neutron Flux scram instrumentation required in proposed ITS 3.3.1.1, Reactor Protection System Instrumentation." Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference into the UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. Additionally, a discussion of the scoop tube stop settings and verification requirements will be included in the UFSAR, with changes controlled by the provisions of 10 CFR 50.59.

LA.3 The CTS 3.6.A Action 2 requirement to "immediately initiate measures to place the unit in at least STARTUP" when no recirculation loops are in operation is relocated to the Bases in the form of a discussion that "action must be taken as soon as practicable" to be in MODE 2. Immediate action may not always be the conservative method to assure safety. The 8 hour Completion Time of ITS 3.4.1 Required Action A.1 ensure appropriate actions are taken in a timely manner to place the unit in MODE 2. Therefore, the relocated requirement 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.

"Specific"

L.1 The explicit requirement in CTS 3.6.A Action 1.e to electrically prohibit the idle recirculation pump from starting except to permit testing in preparation for returning the pump to service has been deleted. This requirement is not necessary to minimize the consequences of any design basis accident. Plant operating practice and procedures are adequate to ensure the pumps are not inadvertently started. In addition, the requirements in CTS 3.6.D (ITS 3.4.9, "Reactor Coolant System Pressure and Temperature (P/T) Limits") will help ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances.

L.2 The required action of CTS 3.6.C Action 2 to trip one of the recirculation pumps when the speed mismatch (i.e. flow mismatch) is not within limits has been deleted. It has been replaced with a requirement (ITS 3.4.1 ACTION B) to declare the loop with the low flow "not in operation." Once the declaration has been made, the appropriate actions for single loop operation must be taken in accordance with CTS 3.6.A.1 (ITS 3.4.1). While a shutdown of the loop may be preferred under some conditions, declaring a pump not in operation will ensure the proper actions are taken in accordance with the single loop analysis.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.3 CTS 4.6.C requires the recirculation pump speed mismatch (i.e., jet pump loop flow mismatch in ITS) to be verified within the limits once per 24 hours when in Operational MODES 1 and 2 during two recirculation loop operation. CTS 4.0.D requires the Surveillances to be met prior to entry into the applicable Mode or other specified conditions. CTS 4.6.C 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 added (proposed SR 3.4.1.1 Note) providing an allowance for time to initiate and complete the Surveillance to avoid intentional entry into the ACTIONS each time the second recirculation pump is started. The time allowed is consistent with the current frequency of the Surveillance (24 hours), and is therefore considered acceptable.

L.4 CTS 3.6.C requires the recirculation pump speeds to be maintained within prescribed limits. With THERMAL POWER ≥ 80% of RATED THERMAL POWER the recirculation pump speeds must be within 10% of each other, and with THERMAL POWER < 80% of RATED THERMAL POWER, recirculation pump speeds must be within 15% of each other. In proposed SR 3.4.1.1, the jet pump loop flow mismatch with both recirculation loops in operation is: ≤ 10% of rated core flow when operating at < 70% of rated core flow; and ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow. The recirculation loop mismatch criteria has been changed from a recirculation pump speed comparison to a core flow comparison. In addition, the cutoff point for the criteria is with respect to total core flow instead of thermal power level. The proposed mismatch tolerance is actually smaller than in CTS at high pump speeds and larger than in CTS at lower pump speeds, based on the relationship between jet pump loop flow and recirculation pump speed. Although, the change is actually more restrictive at higher pump speeds, it is less restrictive on plant operation since the tolerance is larger at lower pump speeds. This change is acceptable since the plant normally operates at high pump speeds where the tolerance has been decreased. The cutoff point (percent of rated core flow instead of RATED THERMAL POWER) has been changed since the capability to match the recirculation loops are influenced more by core flow than by THERMAL POWER which can be influenced by both withdrawing control rods or increasing core flow. The overall change is acceptable because the proposed values are consistent with the loss of coolant accident (LOCA) analysis and a small mismatch has been determined to be acceptable based on engineering judgement. C

RELOCATED SPECIFICATIONS

None

A.1

A.2 — <general organization>

3.6 - LIMITING CONDITIONS FOR OPERATION

F. Relief Valves

LCO 3.4.3 5 reactor coolant system relief valves ~~and the reactuation time delay of two relief valves~~ shall be OPERABLE with the following settings:

<See ITS 3.3.6.3>

Relief Function Setpoint (psig)
Open
≤ 1112 psig
≤ 1112 psig
≤ 1135 psig
≤ 1135 psig
≤ 1135 psig[a]

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. ~~With one or more relief valves open, provided that suppression pool average water temperature is <110°F, take action to close the open relief valve(s); if suppression pool average water temperature is ≥110°F place the reactor mode switch in the Shutdown position.~~ — L.1

4.6 - SURVEILLANCE REQUIREMENTS

F. Relief Valves

<See ITS 3.3.6.3>

1. The relief valve function and the reactuation time delay function instrumentation shall be demonstrated OPERABLE by performance of a:
   a. CHANNEL FUNCTIONAL TEST of the relief valve function at least once per 18 months, and a
   b. CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST of the entire system at least once per 18 months.

2. ~~Deleted.~~

add proposed SRs 3.4.3.2 and 3.4.3.3 — A.4

a Target Rock combination safety/relief valve. — LA.2

C

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS LCO 3.6.E ensures that the appropriate number of safety valves are available to protect the reactor vessel from overpressure during upset conditions as required by the ASME Boiler and Pressure Vessel Code. Proposed LCO 3.6.E (see Discussion of Change A.5) excludes the Target Rock combination safety and relief valve from the safety valve function OPERABILITY requirements of the LCO, but does not specify the number of safety valve functions (safety valves) that are required to be OPERABLE. The number of required safety valves is determined from plant controlled documents. However, the NUREG-1433 presentation of CTS LCO 3.6.E (ISTS LCO 3.4.3) specifies the number (in brackets) of safety valves required to be OPERABLE in order to satisfy the LCO. Therefore, proposed ITS LCO 3.4.3 includes the plant specific requirement that 8 safety valves shall be OPERABLE. Since this change proposes to include a specific number of required safety valves in the ITS, the number of valves will no longer be controlled by ComEd, subject to the provisions of 10 CFR 50.59. Instead, the number of required safety valves will be controlled by the NRC, pursuant to 10 CFR 50.90. As such, this change represents an additional restriction on plant operation and is considered a more restrictive change.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.6.E footnote (a), relating to lift setting pressure of the safety valves (the lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures), are proposed to be relocated to the Bases. The requirements of proposed SR 3.4.3.1 are adequate to ensure safety valve lift setpoints are within required settings. As a result, the details relocated to the Bases are not necessary for ensuring safety valve setpoints are maintained within required settings and do not need 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.

LA.2 The detail in CTS 3.6.F LCO footnote (a) that one of the relief valves is a "Target Rock" combination safety/relief valve is proposed to be relocated to the Bases. Proposed LCO 3.4.3 will continue to require 5 relief valves to be OPERABLE. In addition, the Bases provides sufficient description of both the relief valves and the safety/relief valve. This detail is not necessary to ensure lift C

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 (cont'd) settings are maintained properly. As such, this 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.

LA.3 The testing requirements of CTS 4.6.E.2 for safety valve setting verification are proposed to be relocated to the Inservice Testing (IST) Program. These testing requirements demonstrate the Reactor Coolant System (RCS) safety valves are OPERABLE. However, the IST Program, required by 10 CFR 50.55a, provides requirements for the testing of all ASME Code Class 1, 2, and 3 valves in accordance with applicable codes, standards, and relief requests and is endorsed by the NRC for Dresden 2 and 3. Compliance with 10 CFR 50.55a, and as a result the IST Program and implementing procedures, is required by the Dresden 2 and 3 Operating Licenses. These controls are adequate to ensure the required testing to demonstrate OPERABILITY is performed. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the relocated requirements in the IST Program will be controlled by the provisions of 10 CFR 50.55a.

"Specific"

L.1 CTS 3.6.F Action 1 requires an open relief valve to be closed provided the suppression pool temperature is < 110°F. If unable to close the open relief valve, or if suppression pool temperature is ≥ 110°F, the reactor mode switch must be placed in shutdown. This Action is not included in the ITS. Required Actions for open relief valves are implicit in the Actions of CTS 3.7.K and ITS 3.6.2.1. Required Action D.1 of ITS 3.6.2.1 will also require that the reactor mode switch be immediately placed in shutdown if the suppression pool average temperature is ≥ 110°F. Action 1 of CTS 3.6.F is anticipatory of this requirement in the event of an open relief valve, and preemptive in all cases. This Action represents detailed methods of responding to an event and not necessarily a compensatory action for failure to meet this LCO. As such it is not appropriate for the ITS and is adequately addressed in Dresden 2 and 3 Emergency Operating Procedures and by ITS 3.6.2.1, the Suppression Pool Temperature LCO. Therefore, CTS 3.6.F, Action 1 is proposed to be deleted from Technical Specifications.

RELOCATED SPECIFICATIONS

None

A.1

PRIMARY SYSTEM BOUNDARY

3.6 - LIMITING CONDITIONS FOR OPERATION

3. Nuclear Heatup and Cooldown:

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Region as shown on Figure 3.6.K-~~5~~ 3, and (SR 3.4.9.2) A.9

   b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour. (SR 3.4.9.1)

4. The reactor vessel flange and head flange temperature ≥83°F when reactor vessel head bolting studs are under tension. (SR 3.4.9.5, SR 3.4.9.6, SR 3.4.9.7)

4.6 - SURVEILLANCE REQUIREMENTS

4. The reactor vessel flange and head flange temperature shall be verified to be ≥83°F:

   a. In OPERATIONAL MODE 4 when the reactor coolant temperature is:

   add proposed SR 3.4.9.7 Note | add proposed SR 3.4.9.6 Note | A.5

   1) ≤113°F, at least once per 12 hours. (SR 3.4.9.7)

   2) ≤93°F, at least once per 30 minutes. (SR 3.4.9.6)

   b. ~~Within 30 minutes prior to and~~ at least once per 30 minutes during tensioning of the reactor vessel head bolting studs. (SR 3.4.9.5) A.6

APPLICABILITY:

At all times.

ACTION:

add proposed Conditions A and C Notes A.2

With any of the above limits exceeded,

(ACTIONS A and C)

1. Restore the reactor vessel metal temperature and/or pressure to within the limits within 30 minutes ~~without exceeding the applicable primary system coolant temperature rate of change limit,~~ and A.3 A.10 C

2. ~~Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system and~~ determine that the reactor coolant system remains acceptable for continued operations within 72 hours, or LA.1 L.2

(ACTION B)

3. Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

ADMINISTRATIVE

A.9 (cont'd) reactor pressure vessel of each unit to a maximum of 32 Effective Full Power Years. The proposed changes rely on recently approved American Society of Mechanical Engineers methodology for determining allowable pressure and temperature limits. A similar Technical Specifications amendment was recently issued for Duke Energy, Oconee Nuclear Station. As such, this change is administrative.

A.10 The CTS 3.6.K Action 1 detail that the applicable primary system coolant temperature rate of change limit cannot be exceeded while restoring the reactor vessel metal temperature and/or pressure to within the limits has been deleted. CTS LCO 3.0.A (ITS LCO 3.0.1) requires compliance with the Limiting Conditions for Operation during the Operational Modes or other conditions specified. Since the primary system coolant temperature rate of change limit is specified in CTS LCO 3.6.K (ITS LCO 3.4.9), this added detail is not necessary and its removal is considered administrative. C

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS LCO 3.6.D establishes the conditions for startup of an idle recirculation loop. The temperature limitations are not currently specified in the LCO since they are specified in the Dresden Administrative Technical Requirements (DATR) manual. As discussed in Discussion of Change A.7 above, the CTS 3.6.D requirements have been combined into the RCS P/T Limits Specification (ITS 3.4.9). As such, proposed ITS SRs 3.4.9.3 and 3.4.9.4 verify the temperature limitations for the startup of an idle loop have been met prior to starting the idle loop recirculation pump. The BWR ISTS, NUREG-1433, Revision 1, presentation of these SRs (NUREG SRs 3.4.10.3 and 3.4.10.4) references the Pressure and Temperature Limits Report (PTLR) to obtain the temperature limit values. Since a PTLR has not been approved by the NRC for Dresden 2 and 3, the proposed presentation of ITS SRs 3.4.9.3 and 3.4.9.4 removes references to the PTLR and includes the specific limit values as specified in the DATR. Since this change proposes to include specific limit values in the ITS, the limits will no longer be administratively controlled by ComEd, subject to the provisions of 10 CFR 50.59. Instead, the limits will be controlled by the NRC, pursuant to 10 CFR 50.90. As such, this change represents as additional restriction on plant operation and is considered a more restrictive change.

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.2 The CTS 3.6.D footnote a allowance that the differential temperature between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is not applicable below 25 psig reactor pressure has been deleted. Therefore, ITS SR 3.4.9.3 will require the differential temperature requirement between the reactor pressure vessel coolant and the bottom head coolant to be within limits ($< 145°F$) in MODES 1, 2, 3, and 4 during a recirculation pump startup. Since, the limit must be met at any reactor pressure in these MODES, this change is more restrictive. This change is necessary to minimize thermal stresses resulting from the startup of an idle recirculation pump.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details in CTS 3.6.K Action 2 to perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System is proposed to be relocated to the Bases. The requirements in proposed ITS 3.4.9 Required Action A.2 and C.2 to determine RCS is acceptable for continued operation and the Condition A and C Note that the applicable action shall be completed if this Condition is entered ensures the current requirement is met. In addition, the Bases for these Required Actions indicates that an engineering evaluation shall be performed. As such, 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.

LA.2 The details of the CTS 3.6.D Action and CTS 4.6.D relating to operational limits (loop flow) during a return to two recirculation pump operation from single recirculation loop operation are proposed to be relocated to the UFSAR. The single loop flow rate is considered an operational limit since it is not directly related to the ability of the system to perform its safety analysis functions. The flow rate is limited only to restrict reactor vessel internals vibration to within acceptable limits during restart of the second pump. These requirements are oriented toward maintaining long term OPERABILITY of the recirculation loops and do not necessarily have an immediate impact on their OPERABILITY. As such, 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 the 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 CTS 4.6.K.2.a requires the rate of change of primary system coolant temperature to be determined within limits 15 minutes prior to withdrawal of control rods and at least once per 30 minutes during primary system heatup or cooldown. The requirement to verify the rate of change during the 15 minute period prior to withdrawal of control rods has been deleted, however, the Frequency of once every 30 minutes has been retained as proposed in SR 3.4.9.1 during heatup and cooldown. The primary coolant temperature is not expected to change significantly until the reactor becomes critical, therefore, this Surveillance Requirement is not necessary. CTS 4.6.K.2.b, the requirement to verify the reactor vessel metal temperature and pressure to be within the Acceptable Region of the critical core operation curve (CTS Figure 3.6.K-5) once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality, is being retained in ITS SR 3.4.9.2. The proposed Frequencies of proposed SR 3.4.9.1 and 3.4.9.2 are considered acceptable to ensure the RCS P/T limits are met during critical operations. This change is consistent with BWR ISTS, NUREG-1433, Rev. 1.

L.2 CTS 3.6.K Action 2 and the CTS 3.6.D Action specify a Completion Time of 72 hours for the required engineering evaluation with an LCO applicability of "at all times." Proposed ITS 3.4.9, Required Action C.2, (applicable when in conditions other than MODES 1, 2, and 3) requires completion "prior to entering MODE 2 or 3." While Required Action C.2 is intended to be initiated without delay, it is not restricted to a specified Completion Time, only by a restriction on returning to (entering) operating MODES (i.e., 1, 2, or 3) where additional stresses (heatup/criticality) may be imposed. This change is consistent with BWR ISTS, NUREG-1433, Rev. 1, and is considered acceptable since continued plant operation is prohibited until RCS integrity is assured.

RELOCATED SPECIFICATIONS

None

Safety and Relief Valves 1

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<3.6.E> <4.6.E> 3 SR 3.4.3.1	Verify the safety function lift setpoints of the ~~[required]~~ ~~S/RVs~~ are as follows: 5 TSTF-298 change not adopted 1 Safety valves Number of ~~S/RVs~~ / Setpoint (psig) ~~[4]~~ ~~[1090 ± 32.7]~~ ~~[4]~~ ~~[1100 ± 33.0]~~ ~~[3]~~ ~~[1110 ± 33.3]~~ 2: 2 1240 ± 12.4; 2 1250 ± 12.5; 4 1260 ± 12.6 ~~Following testing, lift settings shall be within ± 1%.~~ 4	~~[~~In accordance with the Inservice Testing Program ~~or [18] months]~~ 2
<DOC A.4> SR 3.4.3.2	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each ~~[required]~~ ~~S/RV~~ relief valve opens when manually actuated. 1	~~[18]~~ 24 months ~~[on a STAGGERED TEST BASIS for each valve solenoid]~~ 3 2 3
<DOC A.4> SR 3.4.3.3	---- NOTE ---- Valve actuation may be excluded. -------------- Verify each relief valve actuates on an actual or simulated automatic initiation signal.	24 months 1

C

all changes are [1] unless otherwise indicated

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~8~~ 7 ~~Residual Heat Removal (RHR)~~ Shutdown Cooling [SDC] System—Hot Shutdown

<3.6.D> LCO 3.4.~~8~~ 7 Two ~~RHR shutdown cooling~~ SDC subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one ~~RHR shutdown cooling~~ SDC subsystem shall be in operation.

required

NOTES

1. Both ~~RHR shutdown cooling~~ SDC subsystems and recirculation pumps may ~~be removed from~~ not be in operation for up to 2 hours per 8 hour period. TSTF-153 [4]

2. One required ~~RHR shutdown cooling~~ SDC subsystem may be inoperable for up to 2 hours for the performance of Surveillances. [5]

<Appl 3.6.D> APPLICABILITY: MODE 3, with reactor ~~steam dome pressure~~ vessel coolant temperature ~~≥~~ less than the ~~RHR~~ SDC cut in permissive ~~pressure~~ temperature. [3]

ACTIONS

NOTES

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each ~~RHR shutdown cooling~~ SDC subsystem.

CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.6.D Act 1> A. One or two required ~~RHR shutdown cooling~~ SDC subsystems inoperable. [5]	A.1	Initiate action to restore ~~RHR shutdown cooling~~ SDC subsystem(s) to OPERABLE status.	Immediately
	AND		
			(continued)

all changes are 1 unless otherwise indicated

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.0 Act 1> A. (continued)	A.2 Verify an alternate method of decay heat removal is available for each inoperable ~~RHR shutdown cooling~~ required SDC subsystem. 5	1 hour
	AND	
	A.3 Be in MODE 4.	24 hours
<4.6.0> <3.6.0 Act 2> B. No ~~RHR shutdown cooling~~ required SDC subsystem in operation. AND No recirculation pump in operation.	B.1 Initiate action to restore one ~~RHR shutdown cooling~~ required SDC subsystem or one recirculation pump to operation.	Immediately
	AND	
	B.2 Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND	
	B.3 Monitor reactor coolant temperature and pressure.	Once per hour

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.7 - SHUTDOWN COOLING SYSTEM — HOT SHUTDOWN

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. ISTS 3.4.8 is renumbered as ITS 3.4.7 as a result of the deletion of ISTS 3.4.5, "Reactor Coolant System Pressure Isolation Valve (PIV) Leakage."

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

4. TSTF-153 revised the RHR Shutdown Cooling System-Hot Shutdown LCO (ISTS LCO 3.4.8) Note 1, which provides an exception to the requirement for the required pump to be in operation, to provide a clarification of the intent of the Note consistent with the requirement being excepted. The justification for TSTF-153 described that the change was necessary to eliminate ambiguity that could lead to errors or improper enforcement. However, the change can now lead to a misinterpretation of the allowance of the Note. Specifically, the Note can now be interpreted as <u>requiring</u> the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period, i.e., they must be taken out of operation. The intent of the Note (as described in the associated Bases) is to <u>allow (but not require)</u> the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period. Therefore, the Note is revised to allow the subsystems or pumps to be "not in operation" for up to 2 hours per 8 hour period.

5. The Shutdown Cooling System includes three subsystems, as described in the Background section of the Bases, with only two required to be OPERABLE by the LCO. Therefore, the term "required" was added into the ITS consistent with the use of the word "required" in the ISTS, and the Writer's Guide. C

all changes are 1 unless otherwise indicated

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 8 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown (SDC) 2

<3.6.P> LCO 3.4.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation. SDC required

-----NOTES-----

2 → 1. Both RHR shutdown cooling subsystems and recirculation pumps may ~~be removed from~~ *not be in* operation for up to 2 hours per 8 hour period. TSTF-153 4

3 → 2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances. *required* 5

1. Both required SDC subsystems may be not in operation during hydrostatic testing. 3

<Appl 3.6.P> APPLICABILITY: MODE 4.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.P Act 1> A. One or two *required* RHR shutdown cooling subsystems inoperable. 5	A.1 Verify an alternate method of decay heat removal is available for each inoperable *required* RHR shutdown cooling subsystem. SDC	1 hour AND Once per 24 hours thereafter

(continued)

all changes are 1 unless otherwise indicated

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.P Act 2> <4.6.P> B. ~~No~~ SDC ~~RHR shutdown cooling~~ subsystem in operation. 5 required AND No recirculation pump in operation.	B.1 Verify reactor coolant circulating by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND B.2 Monitor reactor coolant temperature and pressure. 6	Once per hour

C

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<4.6.P> SR 3.4.~~9~~.1 2 8 Verify one SDC ~~RHR shutdown cooling~~ subsystem or recirculation pump is operating.	12 hours

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.8 - SHUTDOWN COOLING SYSTEM — COLD SHUTDOWN

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. ISTS 3.4.9 is renumbered as ITS 3.4.8 as a result of the deletion of ISTS 3.4.5, "Reactor Coolant System Pressure Isolation Valve (PIV) Leakage."

3. Note 1 to ITS LCO 3.4.8 has been added consistent with the current licensing basis. The subsequent Notes have been renumbered to reflect this addition.

4. TSTF-153 revised the RHR Shutdown Cooling System-Cold Shutdown LCO (ISTS LCO 3.4.9) Note 1, which provides an exception to the requirement for the required pump to be in operation, to provide a clarification of the intent of the Note consistent with the requirement being excepted. The justification for TSTF-153 described that the change was necessary to eliminate ambiguity that could lead to errors or improper enforcement. However, the change can now lead to a misinterpretation of the allowance of the Note. Specifically, the Note can now be interpreted as requiring the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period, i.e., they must be taken out of operation. The intent of the Note (as described in the associated Bases) is to allow (but not require) the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period. Therefore, the Note is revised to allow the subsystems or pumps to be "not in operation" for up to 2 hours per 8 hour period.

5. The Shutdown Cooling System includes three subsystems, as described in the Background section of the Bases, with only two required to be OPERABLE by the LCO. Therefore, the term "required" was added into the ITS consistent with the use of the word "required" in the ISTS and the Writer's Guide. C

6. This change was made to be consistent with the current licensing basis and NUREG-1434, ISTS 3.4.10. C

BASES (continued)

SURVEILLANCE REQUIREMENTS

~~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.~~ 1

(5) (1)

SR 3.4.6.2

drywell floor drain sump monitoring system

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. its system 1 2

INSERT SR 3.4.5.1 TSTF -205 C 2

(5) (2) 1

SR 3.4.6.3

1 the drywell floor drain sump monitoring system

SR 3.4.5.2 is based on the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

This SR is for the performance of a CHANNEL CALIBRATION of ~~required leakage detection~~ instrumentation channel~~s~~. ~~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.~~ (i.e., drywell floor drain sump pump discharge flow integrator)

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 30~~. UFSAR, Section 3.1.2.4.1 2

2. Regulatory Guide 1.45, May 1973.

3. ~~FSAR, Section [5.2.7.2.1]~~. "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," 4

(3) 4. GEAP-5620, April 1968.

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

(5) 6. ~~FSAR~~, Section ~~[5.2.7.5.2]~~. 5.2.5.6.4 (U) 5

2

TSTF
-205

Insert SR 3.4.5.1

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.

all changes are 1 unless otherwise indicated

BASES

LCO (continued)

common discharge piping. Thus, to meet the LCO, both pumps in one loop or one pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

is a | It is | is | SDC | 3 | 2 | and recirculation pumps | TSTF-153 | not be in operation | C

Note 1 permits both RHR shutdown cooling subsystems to be shut down for a period of 2 hours in an 8 hour period. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

SDC | required

APPLICABILITY

In MODE 3 with reactor steam dome pressure below the RHR cut in permissive pressure (i.e., the actual pressure at which the interlock resets) the RHR System may be operated in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature. Otherwise, a recirculation pump is required to be in operation.

5 | temperature | vessel coolant temperature | SDC | temperature | SDC

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut in permissive pressure is typically accomplished by condensing

vessel coolant temperature | temperature | SDC | 5 | temperatures

(continued)

allchanges are [1] unless otherwise indicated

BASES

ACTIONS

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

without delay. In this condition, the remaining OPERABLE subsystem can provide the necessary decay heat removal. The overall reliability is reduced, however, because a single failure in the OPERABLE subsystem could result in reduced ~~RHR/shutdown cooling~~ SDC capability. 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~~ required ~~RHR shutdown cooling~~ 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.

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 ~~Spent Fuel Pool Cooling System~~ Condensate/Feed and Main Steam Systems 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). [3]

However, due to the potentially reduced reliability of the alternate methods of decay heat removal, it is also required to reduce the reactor coolant temperature to the point where MODE 4 is entered.

B.1, B.2, and B.3

With no required ~~RHR shutdown cooling~~ SDC subsystem and no recirculation pump in operation, except as permitted by LCO Note 1, ~~reactor coolant circulation by the RHR shutdown cooling~~ SDC subsystem or recirculation pump must be restored without delay. [C]

Until ~~RHR~~ 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

(continued)

all changes are 1 unless otherwise indicated

BASES

LCO (continued)

~~in one loop of one pump in each of the~~ two loops is a must be OPERABLE. Since the piping ~~and heat exchangers are~~ it is passive components is that ~~are~~ assumed not to fail, ~~they are~~ allowed to be common to both subsystems. ~~In MODE 4, the RHR cross tie valve (2E11-F010) may be opened to allow pumps in one loop to discharge through the opposite recirculation loop to make a complete subsystem.~~ 3 Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one SDC ~~RHR shutdown cooling~~ subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. 2 ~~However,~~ To ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. and recirculation pumps required

Insert Note 1

Note ~~1~~ 2 permits both SDC ~~RHR shutdown cooling~~ subsystems and recirculation pumps to ~~be shut down~~ not be in operation for a period of 2 hours in an 8 hour period. TSTF-53 Note ~~2~~ 3 allows one SDC ~~RHR shutdown cooling~~ subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected ~~RHR~~ System or on some other plant system or component that necessitates placing the ~~RHR~~ System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the ~~RHR~~ subsystems or other operations requiring ~~RHR~~ flow interruption and loss of redundancy. required C

APPLICABILITY

In MODE 4, the SDC ~~RHR Shutdown Cooling~~ System ~~may~~ must be OPERABLE and one SDC subsystem shall 5 be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below ~~200~~ 212°F. Otherwise, a recirculation pump is required to be in operation. 3

In MODES 1 and 2, and in MODE 3 with reactor ~~steam dome~~ vessel coolant temperature 2 ~~pressure~~ temperature greater than or equal to the ~~RHR~~ SDC cut in permissive ~~pressure~~ temperature, this LCO is not applicable. Operation of the ~~RHR~~ SDC System in the shutdown cooling mode is not allowed above this ~~pressure~~ temperature because the RCS ~~pressure~~ may exceed the design ~~pressure~~ of the shutdown cooling piping. Decay heat removal at reactor ~~pressures~~ temperatures greater than or equal to the ~~RHR~~ SDC cut in permissive ~~pressure~~ temperature is typically accomplished by condensing 2

(continued)

1 INSERT NOTE 1

Note 1 allows both required SDC subsystems to not be in operation during hydrostatic testing. This is acceptable since adequate reactor coolant circulation will be achieved by operation of a reactor recirculation pump and since systems are available to control reactor coolant temperature.

all changes are 1 unless otherwise indicated

BASES

ACTIONS

A.1 (continued)

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 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 ~~Spent Fuel Pool Cooling System~~ and the Reactor Water Cleanup System.

3 Condensate/Feed and Main Steam 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 ~~RHR shutdown cooling~~ SDC subsystem and no recirculation pump in operation, except as permitted by LCO Note ~~1~~ 5 and 2, and until ~~RHR~~ 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 ~~RHR Shutdown Cooling~~ 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)

L.4 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 recirculation loop speed mismatch criteria has been changed from a recirculation pump speed comparison to a core flow comparison. In addition, the cutoff point for the criteria is with respect to total core flow instead of thermal power level. The speed of the recirculation pumps and the jet pump loop flows are not considered to be an initiator of an analyzed event, therefore this change will not increase the probability of the event. The change to the recirculation loop mismatch criteria is consistent with the limits assumed by the loss of coolant accident (LOCA) analysis. Therefore, the proposed change will not increase the probability or consequences of any accident previously evaluated. C

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

   The proposed changes do not involve any design changes, plant modifications, or changes in plant operation. The system will continue to be operated and function in the same way as before the change. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed mismatch criteria will ensure an event will be bounded by the safety analysis.

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 less than or equal to the acceptable $A/\sqrt{k}$ design value of 0.18 ft^2. However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion is $\leq$ 2% of the acceptable $A/\sqrt{k}$ design value of 0.18 ft^2.	24 months AND ------NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass -------------- 12 months

C

BASES

SURVEILLANCE REQUIREMENTS

<u>SR 3.6.1.1.2</u> (continued)

test, risk of high radiation exposure, and the remote possibility of a component failure that is not identified by some other primary containment SR. 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.

BACKGROUND (continued)

maximum negative containment (drywell and suppression chamber) pressure to within design limits. The maximum depressurization rate is a function of the primary containment spray flow rate and temperature and the assumed initial conditions of the primary containment atmosphere. Low spray temperatures and atmospheric conditions that yield the minimum amount of contained noncondensible gases are assumed for conservatism.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the reactor building-to-suppression chamber vacuum breakers are presented in Reference 1 as part of the accident response of the containment systems. Internal (suppression-chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls, which form part of the primary containment boundary.

The safety analyses assume the external vacuum breakers to be closed initially, with the mechanical vacuum breakers counter balanced to open at 0.5 psid and to be fully open in 5 seconds. The air operated butterfly valve vacuum breakers are assumed to open concurrent with the mechanical vacuum breakers and be full open in 30 seconds (Ref. 1). Since only one of the two parallel 20 inch vacuum breaker lines is required to protect the suppression chamber from excessive negative differential pressure, the single active failure criterion is satisfied. Design Basis Accident (DBA) analyses assume the vacuum breakers to be closed initially and that at least one vacuum breaker in each line remains closed and leak tight with positive primary containment pressure.

Three cases were considered in the safety analyses to determine the adequacy of the external vacuum breakers:

a. Inadvertent actuation of one drywell spray loop during normal operation;

b. Inadvertent actuation of one drywell spray loop during normal operation with the suppression chamber free air volume completely filled with saturated steam; and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Inadvertent actuation of one drywell spray loop during normal operation with the suppression chamber free air volume filled with an air/steam mixture.

The results of these three cases show that the external vacuum breakers, with an opening setpoint of 0.5 psid, are capable of maintaining the differential pressure within design limits.

The reactor building-to-suppression chamber vacuum breakers satisfy 10 CFR 50.36(c)(2)(ii).

LCO

All reactor building-to-suppression chamber vacuum breakers are required to be OPERABLE to satisfy the assumptions used in the safety analyses. The requirement ensures that the two vacuum breakers (mechanical vacuum breaker and air operated butterfly valve) in each of the two lines from the reactor building to the suppression chamber airspace are closed (except during testing or when performing their intended function). Also, the requirement ensures both vacuum breakers in each line will open to relieve a negative pressure in the suppression chamber.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture, which purges the drywell of air and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell, which, after the suppression chamber-to-drywell vacuum breakers open (due to excessive differential pressure between the suppression chamber and drywell), would result in depressurization of the suppression chamber. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3. Excessive negative pressure inside primary containment could occur due to inadvertent initiation of drywell sprays.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining reactor building-to-suppression chamber vacuum breakers OPERABLE is not required in MODE 4 or 5.

(continued)

BASES (continued)

ACTIONS

A Note has been added to provide clarification that, for the purpose of this LCO, separate Condition entry is allowed for each reactor building-to-suppression chamber vacuum breaker line.

A.1

With one or more lines with one vacuum breaker not closed, the leak tight primary containment boundary may be threatened. Therefore, the inoperable vacuum breakers must be restored to OPERABLE status or the open vacuum breaker closed within 7 days. The 7 day Completion Time takes into account the redundancy capability afforded by the remaining breakers, the fact that the OPERABLE breaker in each of the lines is closed, and the low probability of an event occurring that would require the vacuum breakers to be OPERABLE during this period.

B.1

With one or more lines with two vacuum breakers not closed, primary containment integrity is not maintained. Therefore, one open vacuum breaker must be closed within 1 hour. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, "Primary Containment," which requires that primary containment be restored to OPERABLE status within 1 hour.

C.1

With one line with one or more vacuum breakers inoperable for opening, the leak tight primary containment boundary is intact. The ability to mitigate an event that causes a containment depressurization is threatened, however, if both vacuum breakers in at least one vacuum breaker penetration are not OPERABLE. Therefore, the inoperable vacuum breaker must be restored to OPERABLE status within 7 days. This is consistent with the Completion Time for Condition A and the fact that the leak tight primary containment boundary is being maintained.

(continued)

BASES

ACTIONS (continued)

D.1

With two lines with one or more vacuum breakers inoperable for opening, the primary containment boundary is intact. However, in the event of a containment depressurization, the function of the vacuum breakers is lost. Therefore, all vacuum breakers in one line must be restored to OPERABLE status within 1 hour. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, which requires that primary containment be restored to OPERABLE status within 1 hour.

E.1 and E.2

If any Required Action and associated Completion time can not 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 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.1

Each vacuum breaker is verified to be closed to ensure that a potential breach in the primary containment boundary is not present. This Surveillance is performed by observing local or control room indications of vacuum breaker position. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

Two Notes are added to this SR. The first Note allows reactor-to-suppression chamber vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers. The second Note is included to clarify that vacuum breakers open due to an actual differential pressure are not considered as failing this SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.7.2

Each vacuum breaker must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This ensures that the safety analysis assumptions are valid. The 92 day Frequency of this SR was developed based upon Inservice Testing Program requirements to perform valve testing at least once every 92 days.

SR 3.6.1.7.3

Demonstration of vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of $\leq$ 0.5 psid is valid. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. For this plant, the 24 month Frequency has been shown to be acceptable, based on operating experience, and is further justified because of other surveillances performed at shorter Frequencies that convey the proper functioning status of each vacuum breaker.

REFERENCES

1. UFSAR, Section 6.2.1.2.4.

BASES (continued)

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 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.

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

Movement of irradiated fuel assemblies in the secondary containment, CORE ALTERATIONS, and OPDRVs can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

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 C

(continued)

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

Verifying that one secondary containment access door in each access opening is 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. 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

(continued)

ACTIONS

B.1 (continued)

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths with two isolation valves. This clarifies that only Condition A is entered if one SCIV is inoperable in each of two penetrations.

C.1 and C.2

If any Required Action and associated Completion Time 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 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.

D.1, D.2, and D.3

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action D.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 fuel 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies that each secondary containment manual isolation valve and blind flange that is not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

Since these SCIVs are readily accessible to personnel during normal operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

Two Notes have been added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these SCIVs, once they have been verified to be in the proper position, is low.

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.2.2

Verifying that the isolation time of each power operated, automatic SCIV is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that assumed in the safety analyses. The Frequency of this SR is 92 days.

SR 3.6.4.2.3

Verifying that each automatic SCIV closes on a secondary containment isolation signal is required to prevent leakage of radioactive material from secondary containment following a DBA or other accidents. This SR ensures that each automatic SCIV will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 15.6.5.
2. UFSAR, Section 15.7.3.
3. Technical Requirements Manual.

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation will occur, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must immediately be suspended. Suspension of these activities must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

The Required Actions of Condition C 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. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

D.1

If both SGTS subsystems are inoperable in MODE 1, 2, or 3, the SGT system may not be capable of supporting the required radioactivity release control function. Therefore, one SGT subsystem must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of supporting the required radioactivity release control function in MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring the SGT System) occurring during periods where the required radioactivity release control function may not be maintained is minimal.

(continued)

ACTIONS (continued)

E.1 and E.2

If one SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, 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 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.

F.1, F.2, and F.3

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

Required Action F.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.

(continued)

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.

LD.1

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 ~~9~~ 12 months until two consecutive tests meet the specified limit, at which time the ~~18~~ 24 month test schedule may be resumed.

L.3

L.2

LD.1

C

ADMINISTRATIVE

A.6 (cont'd) 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"

LD.1 The Frequency for performing CTS 4.7.K.5 (proposed SR 3.6.1.1.2), 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 required if a routine test fails two times in a row, to facilitate a change to the Dresden 2 and 3 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.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. This SR ensures that the boundary between the drywell airspace and the suppression chamber airspace is maintained to ensure the pressure suppression function is OPERABLE by limiting the amount of bypass steam leakage which would not be directed through the suppression pool water. The suppression chamber-to-drywell vacuum breakers are the only active mechanical devices in the boundary between the drywell air space and the suppression chamber and are functionally tested on a more frequent basis by ITS SR 3.6.1.8.2 to ensure their OPERABILITY. In addition, ITS SR 3.6.1.8.1 verifies the suppression chamber-to-drywell vacuum breakers are closed every 14 days. Although the more frequent tests do not directly ensure the leak tightness of the drywell to suppression chamber boundary, they do ensure the valves are functional and closed. Based on the passive design of the

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) suppression chamber-to-drywell vacuum breakers and the more frequent functional testing of the suppression chamber-to-drywell vacuum breakers, the impact, if any, from this change on component and system availability is minimal.

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 4.7.k.5 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.2 above), primary containment structural integrity or leakage rates discovered outside acceptance criteria (ITS SR 3.6.1.1.1) or the drywell-to-suppression chamber bypass leakage outside limits (ITS SR 3.6.1.1.2) 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 or structural integrity problem is not corrected within 1 hour. With drywell-to-suppression chamber bypass leakage outside of limits in MODE 1, 2, or 3, CTS 3.7.K does not provide actions. Since drywell-to-suppression chamber leakage is an attribute 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 current Actions allowed for structural integrity and primary containment leakage not within limits in CTS 3.7.A. 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 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.

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.

L.3 CTS 3.7.K.3 requires the 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. In addition, CTS 4.7.K.5 requires this test to be performed every 18 months (extended to 24 months in accordance with Discussion of Change LD.1). ITS SR 3.6.1.1.2 requires the drywell-to-suppression chamber bypass leakage to be less than or equal to the bypass leakage limit. The bypass leakage limit is specified to be less than or equal to the acceptable $A/\sqrt{k}$ design value of 0.18 ft^2. However, ITS SR 3.6.1.1.2 further requires the drywell-to-suppression chamber bypass leakage to be $\leq 2\%$ of the acceptable $A/\sqrt{k}$ design value during the first unit startup following bypass leakage testing performed in accordance with proposed SR 3.6.1.1.2. The current bypass leakage limit (equivalent leakage through a 1 inch diameter orifice) is equivalent to the proposed bypass leakage required during testing ($\leq 2\%$ of the acceptable $A/\sqrt{k}$ design value) as documented in Dresden Station Special Report No. 23 submitted to D.J. Skovholt (NRC) from L. D. Butterfield (Commonwealth Edison) on April 23, 1973. Proposed SR 3.6.1.1.2 is consistent with the current drywell-to-suppression chamber leakage rate limit testing requirements described in the CTS 3.7.K.3, with two exceptions. Proposed SR 3.6.1.1.2 will continue to require that drywell-to-suppression chamber bypass leakage be less than or equal to 2% of the acceptable $A/\sqrt{k}$ design value (equivalent leakage through a 1 inch diameter orifice) during the first unit startup following bypass leakage testing performed in accordance with ITS 3.6.1.1, however, bypass leakage will be considered to be acceptable if it is less than or equal to the design $A/\sqrt{k}$ leakage limit at all other times between required tests.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd) This change to CTS 3.7.K.3 is considered acceptable based upon a history of satisfactory results from prior drywell-to-suppression chamber bypass leakage rate testing. The second exception is that the detail of the initial differential pressure to perform the test has been deleted from the Technical Specifications. These details for testing are not necessary in the Technical Specifications since the proposed limits will ensure that the leakage limits will be met during plant operations.

C

RELOCATED SPECIFICATIONS

None

A.1

3.7 - LIMITING CONDITIONS FOR OPERATION

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

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

ACTION:

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.

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.

see ITS 3.6.4.1

4.7 - SURVEILLANCE REQUIREMENTS

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

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. At least one door in each secondary containment air lock is closed. L.2 B

   add Proposed Required Action A.1

   Required Action A.2 and SR 3.6.4.2.1

   b. All secondary containment penetrations[a] not capable of being closed by OPERABLE secondary containment automatic isolation ~~dampers~~ valves and required to be closed during accident conditions are closed. M.2 A.2 C

   Not locked, sealed, or otherwise secured L.5

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.

SR 3.6.4.2.1 Note 1 and Required Action A.2 Note

* 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. L.1

SR 3.6.4.2.1 Note 2

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 ITS SR 3.6.4.2.2 has been added to the secondary containment isolation damper Surveillance Requirements specified in CTS 4.7.O. ITS SR 3.6.4.2.2 requires the isolation time of each power operated, automatic SCIV to be verified within limits. The satisfactory completion of this SR provides assurance that the secondary containment isolation valves will function and the secondary containment will perform as assumed in the safety analyses. The proposed Frequency of ITS SR 3.6.4.2.2 is 92 days, which is consistent with the Frequency for the stroke time testing requirements of the Inservice Testing Program. This Frequency is also consistent with the isolation time verification requirements for power operated, automatic PCIVs (ITS SR 3.6.1.3.5 and CTS 4.7.D.3). The addition of this new SR and its performance in accordance with the proposed Frequency is a restriction on plant operation.

M.2 CTS 4.7.N.2.b requires all secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions to be closed. This can be met by a single manual valve being closed. CTS 3.7.O requires each secondary containment ventilation system automatic isolation damper to be OPERABLE. CTS 3/4.7.O does not prescribe limitations on manual valves. ITS LCO 3.6.4.2 requires each SCIV to be OPERABLE and proposed SR 3.6.4.2.1 requires the verification that each secondary containment isolation manual valve and blind flange that is not locked, sealed or otherwise secured and is required to be closed during accident conditions is closed. In the ITS, the SCIVs include both the automatic isolation dampers as well as secondary containment manual isolation valves. Since some penetration flow paths include more than one manual isolation valve, this change is more restrictive on plant operation. This change is necessary to ensure the position of all secondary containment isolation valves and blind flanges are properly controlled to ensure design basis assumptions are met. C

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LD.1 The Frequency for performing CTS 4.7.O.2 has been extended from 18 months to 24 months in proposed SR 3.6.4.2.3 to facilitate a change to the Dresden 2 and 3 refuel cycle from 18 months to 24 months. 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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.

SR 3.6.4.2.3 verifies each automatic secondary containment isolation valve (SCIV) actuates to the isolation position on an actual or simulated automatic isolation signal. This is required to prevent leakage of radioactive material from secondary containment following a DBA or other accidents. Extending the Surveillance interval for this verification is acceptable in part because the valves are operated more frequently every 92 days to satisfy the requirements of SR 3.6.4.2.2, which verifies isolation times are within limits. These tests will detect significant failures affecting valve operation that would be detected by conducting the 24 month surveillance test. In addition, the Secondary Containment Isolation system active components and power supplies are designed with redundancy to meet the single active failure criteria, which will ensure system availability in the event of a failure of one of the system components. Also the actual or simulated isolation signal overlaps Logic System Functional Testing performed in SR 3.3.6.2.4 of Secondary Containment Isolation Instrumentation. 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 redundancy and the above discussion, it is concluded that the impact, if any, on system availability is minimal as a result of the change to the SCIV test intervals.

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) Reviews of historical maintenance and surveillance data have shown that this test normally passes the Surveillance 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.

"Specific"

L.1 An allowance is proposed for intermittently opening closed secondary containment isolation valves under administrative control, other than those currently allowed to be opened using CTS 4.7.N, footnote a (locked or sealed-closed penetrations). This is equivalent to the allowance in the existing primary containment Technical Specifications for locked or sealed-closed valves (CTS 3.7.D) and in ITS 3.6.1.3. The administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. The allowance is presented in ITS 3.6.4.2 ACTIONS Note 1 and SR 3.6.4.2.1 Note 2. Opening of secondary containment penetrations on a intermittent basis is required for many of the same reasons as primary containment penetrations and the potential impact on consequences is less significant. The proposed allowance is acceptable due to the low probability of an event that would release radioactivity in the secondary containment during the short time in which the SCIV is open and the administrative controls established to ensure the affected penetration can be isolated when a need for secondary containment isolation is indicated.

L.2 In the event both dampers in a penetration are inoperable in an open penetration, the CTS 3.7.O Action, which requires maintaining one isolation damper OPERABLE, would not be met and an immediate shutdown would be required. ITS 3.6.4.2 ACTION B provides 4 hours prior to commencing a required shutdown. This proposed 4 hour period is consistent with the existing time allowed for conditions when the secondary containment is inoperable. In the event a valve or blind flange is inoperable in a single valve/blind flange penetration, CTS 4.7.N.2.b would not be met, requiring CTS 3.7.N Action 1 or 2 to be entered as appropriate. CTS 3.7.N Action 1 requires the valve/blind flange to be restored within 4 hours or to shutdown the unit, and CTS 3.7.N Action 2 requires immediate suspension of various shutdown evolutions. ITS 3.6.4.2 Required Action A.1 provides 8 hours to commence the unit

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 (cont'd) shutdown or suspend various shutdown evolutions. The proposed change will provide consistency in ACTIONS for these various secondary containment degradations. These changes to CTS 3.7.O are acceptable due to the low probability of an event requiring the secondary containment during the short time in which continued operation is allowed and the capability to isolate a secondary containment penetration is lost. In addition, the penetrations affected by the proposed 8 hour time period are of a small diameter, thus their impact on the secondary containment is not as great as the automatic isolation dampers. B B B

L.3 CTS 4.7.O.1 is proposed to be deleted. Any time the OPERABILITY of a system or component has been affected by repair, maintenance, or replacement of a component, post maintenance testing is required to demonstrate OPERABILITY of the system or component. After restoration of a component that caused a required SR to be failed, ITS SR 3.0.1 requires the appropriate SRs (in this case SR 3.6.4.2.2) to be performed to demonstrate the OPERABILITY of the affected components. Therefore, explicit post maintenance Surveillance Requirements in CTS 4.7.O.1 are not required and have been deleted from the Technical Specifications.

L.4 The phrase "actual or," in reference to the isolation test signal in CTS 4.7.O.2, has been added to proposed SR 3.6.4.2.3, which verifies that each SCIV actuates on an automatic isolation signal. This allows satisfactory automatic SCIV isolations for other than Surveillance purposes to be used to fulfill the Surveillance Requirement. Operability is adequately demonstrated in either case since the SCIV itself cannot discriminate between "actual" or "test" signals.

L.5 The requirements of CTS 4.7.N.2.b, related to verification of the position of secondary containment isolation penetrations not capable of being closed by OPERABLE secondary containment isolation valves (SCIVs), are revised in proposed SR 3.6.4.2.1 and ITS 3.6.4.2 Required Action A.2 (Note 2) to exclude verification of manual valves and blind flanges that are locked, sealed, or otherwise secured in the correct position. The purpose of CTS 4.7.N.2.b is to ensure that manual secondary containment isolation devices that may be misaligned are in correct position to help ensure that post accident leakage of radioactive fluids or gases outside the secondary containment boundary is within design and analysis limits. For manual valves or blind flanges that are locked, sealed or otherwise secured in the correct position, the potential of these devices to be inadvertently misaligned is low. In addition, manual valves and blind flanges that are locked, sealed or otherwise secured in the correct position are

TECHNICAL CHANGES - LESS RESTRICTIVE

L.5 (cont'd) verified to be in the correct position prior to locking, sealing, or securing. As a result of this control of the position of these manual secondary containment isolation devices, the periodic Surveillance of these devices in CTS 4.7.N.2.b is not required to help ensure that post accident leakage of radioactive fluids or gases outside the secondary containment boundary is maintained within the design and analysis limits. This change also provides the benefit of reduced radiation exposure to plant personnel through the elimination of the requirement to check the position of the manual valves and blind flanges, located in the radiation areas, that are locked, sealed or otherwise secured in the correct position.

RELOCATED SPECIFICATIONS

None

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
<4.7A>	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.~~ 1	~~-----NOTE----- SR 3.0.2 is not applicable~~ In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Primary Containment Leakage Rate Testing Program	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] psid~~ bypass leakage is less than or equal to the acceptable A/√K design value of 0.18 ft². However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion is ≤ 2% of the acceptable A/√K design value of 0.18 ft². 2	~~[18]~~ 24 months AND -----NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass -------------- ~~[18]~~ 12 months	3 C 3

$\boxed{3}$ Insert SR 3.6.1.1.2

SR 3.6.1.1.2

The analyses results in Reference 4 are based on a maximum drywell-to-suppression chamber bypass leakage. This Surveillance ensures that the actual bypass leakage is less than or equal to the acceptable A/√k design value of 0.18 ft^2 assumed in the safety analysis. For example, with a typical loss factor of 3 or greater, the maximum allowable leakage area would be approximately 0.3 ft^2, corresponding to a 8-in line size.

As left bypass leakage, prior to the first startup after performing a required bypass leakage test, is required to be ≤ 2% of the drywell-to-suppression chamber bypass leakage limit. At all other times between required leakage rate tests, the acceptance criteria is based on design A/√k. At the design A/√k the containment temperature and pressurization response are bounded by the assumptions of the safety analysis. The leakage test is performed every 24 months, consistent with the difficulty of performing the test, risk of high radiation exposure, and the remote possibility of a component failure that is not identified by some other primary containment SR. 1/C

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. [3] 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. [4] 12

REFERENCES

1. UFSAR, Section 6.2 6.2.1 [4]
2. UFSAR, Section 15.6.5 [1]
3. 10 CFR 50, Appendix J, Option B [1]
4. Dresden Station Special Report No. 23, "Information Concerning Dresden Units 2 and 3 Drywell to Torus Vacuum Breakers," April 1973. [1]

BASES

BACKGROUND (continued)

Low spray temperatures and atmospheric conditions that yield the minimum amount of contained noncondensible gases are assumed for conservatism.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the reactor building-to-suppression chamber vacuum breakers are presented in Reference 1 as part of the accident response of the containment systems. Internal (suppression-chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls, which form part of the primary containment boundary.

Insert 3.6.1.7 ASA 1

The safety analyses assume the external vacuum breakers to be closed initially ~~and to be fully open at [0.5] psid (Ref. 1). Additionally, of the two reactor building-to-suppression chamber vacuum breakers, one is assumed to fail in a closed position to satisfy the single active failure criterion.~~ Design Basis Accident (DBA) analyses ~~require~~ assume the vacuum breakers to be closed initially and ~~to~~ remain closed and leak tight with positive primary containment pressure. that at least one vacuum breaker in each line 5 1

1 Three ~~Five~~ cases were considered in the safety analyses to determine the adequacy of the external vacuum breakers:

~~a. A small break loss of coolant accident followed by actuation of both primary containment spray loops;~~

1 a ~~b~~. Inadvertent actuation of one ~~primary containment~~ drywell 1 spray loop during normal operation;

~~c. Inadvertent actuation of both primary containment spray loops during normal operation;~~

~~d. A postulated DBA assuming Emergency Core Cooling Systems (ECCS) runout flow with a condensation effectiveness of 50%; and~~

~~e. A postulated DBA assuming ECCS runout flow with a condensation effectiveness of 100%.~~

b. Inadvertent actuation of one drywell spray loop during normal operation with the suppression chamber free air volume completely filled with saturated steam; and

c. Inadvertent actuation of one drywell spray loop during normal operation with the suppression chamber free air volume filled with an air/steam mixture.

1 three

The results of these ~~five~~ three cases show that the external vacuum breakers, with an opening setpoint of ~~[~~0.5~~]~~ psid, are 2

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

capable of maintaining the differential pressure within design limits.

The reactor building-to-suppression chamber vacuum breakers satisfy ~~Criterion 3 of the NRC Policy Statement~~. 10CFR 50.36(c)(2)(ii) [1]

LCO

All reactor building-to-suppression chamber vacuum breakers are required to be OPERABLE to satisfy the assumptions used in the safety analyses. The requirement ensures that the two vacuum breakers (~~vacuum breaker~~ mechanical and air operated butterfly valve) in each of the two lines from the reactor building to the suppression chamber airspace are closed (except during testing or when performing their intended function). Also, the requirement ensures both vacuum breakers in each line will open to relieve a negative pressure in the suppression chamber.

APPLICABILITY [4]

In MODES 1, 2, and 3, a DBA could cause pressurization of primary containment. In MODES 1, 2, and 3, the Suppression Pool Spray System is required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside primary containment could occur due to inadvertent initiation of this system. Therefore, the vacuum breakers are required to be OPERABLE in MODES 1, 2, and 3, when the Suppression Pool Spray System is required to be OPERABLE, to mitigate the effects of inadvertent actuation of the Suppression Pool Spray System.

~~Also,~~ In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture, which purges the drywell of air and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell, which, after the suppression chamber-to-drywell vacuum breakers open (due to excessive differential pressure between the suppression chamber and drywell), would result in depressurization of the suppression chamber [1]. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3. drywell sprays. [1]

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining reactor

(continued)

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.7 - REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKERS

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

3. Not used.

4. Inadvertent actuation of the suppression pool spray system is not the main concern for depressurizing the drywell, a LOCA inside the drywell is the main concern. Therefore, this section has been reworded to place proper emphasis on the proper reason. In addition, inadvertent actuation of suppression pool spray is not a concern at all relative to causing an excessive negative pressure event; drywell spray is the system that can cause this event. Therefore the Bases have been changed from suppression pool spray to drywell spray when discussing this event.

5. Changes have been made to reflect those changes made to the Specification.

6. The alternate method has been deleted since it is not valid for Dresden 2 and 3.

7. Editorial change made for enhanced clarity.

8. These changes have been made for consistency with similar phrases in other parts of the Bases and/or to be consistent with the Specification.

BASES

ACTIONS

A.1 (continued)

maintaining [secondary] containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring [secondary] containment OPERABILITY) occurring during periods where [secondary] containment is inoperable is minimal.

B.1 and B.2

If [secondary] containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 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.

1

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

Movement of irradiated fuel assemblies in the [secondary] containment, CORE ALTERATIONS, and OPDRVs can be postulated to cause fission product release to the [secondary] containment. In such cases, the [secondary] containment is the only barrier to release of fission products to the environment. CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended if the [secondary] containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

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

C

C

(continued)

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.

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 ~~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

2

3 one

~~SR 3.6.4.1.2 and~~ SR 3.6.4.1.3

is

3

Verifying that [secondary] containment ~~equipment hatches and~~ access door ~~s~~ ~~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.

in each access opening

3 one

1

TSTF -18

one

INSERT B 3.6.4.1.2

3

(continued)

BASES

ACTIONS

B.1 (continued)

with two isolation valves. This clarifies that only Condition A is entered if one SCIV is inoperable in each of two penetrations.

C.1 and C.2

If any Required Action and associated Completion Time 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 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.

D.1, D.2, and D.3

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the
2 [secondary] containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action D.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 fuel 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

(continued)

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

3

occurred, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing radioactive material to the [secondary] containment, thus placing the plant in a condition that minimizes risk. If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must immediately be suspended. Suspension of these activities must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

2

The Required Actions of Condition C 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. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

C

C

D.1

If both SGTS subsystems are inoperable in MODE 1, 2, or 3, the SGT system may not be capable of supporting the required radioactivity release control function. ~~Therefore, actions are required to enter LCO 3.0.3 immediately.~~

INSERT ACTION D

INSERT ACTION E

5

F

E.1, E.2, and E.3

When two SGT subsystems are inoperable, if applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in [secondary] containment must immediately be suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel

2

3

(continued)

1C

5 Insert ACTION D

Therefore, one SGT subsystem must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of supporting the required radioactivity release control function in MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring the SGT System) occurring during periods where the required radioactivity release control function may not be maintained is minimal.

5 Insert ACTION E

E.1 and E.2

If one SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, 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 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.

BASES

ACTIONS 5 F

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

draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action D.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. F 5 C C

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

(from the control room using the manual initiation switch) 4

Operating each SGT subsystem for ≥ [10] continuous hours ensures that [both] subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation [with the heaters on (automatic heater cycling to maintain temperature)] for ≥ [10] continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system. 2 2

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP. 5 1

(continued)

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. Two DGCW subsystems; and

b. The opposite unit DGCW subsystem capable of supporting its associated diesel generator (DG).

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 associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.2.1 Verify each DGCW subsystem manual 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 an actual or simulated initiation signal.	24 months

BASES (continued)

LCO

The OPERABILITY of the DGCW System is required to provide a coolant source to ensure effective operation of the DGs in the event of an accident or transient. The OPERABILITY of the DGCW System is based on having an OPERABLE pump and an OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring water to the associated DG heat exchangers. The OPERABILITY of the opposite unit's DGCW subsystem is required to provide adequate cooling to ensure effective operation of the required opposite unit's DG heat exchangers in the event of an accident in order to support operation of the shared systems such as the Standby Gas Treatment System and Control Room Emergency Ventilation System.

C

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 be OPERABLE 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.

(continued)

BASES

ACTIONS (continued)

A.1

If one or more DGCW subsystems are inoperable, the associated DG(s) cannot perform their intended function and must be immediately declared inoperable. In accordance with LCO 3.0.6, this also requires entering into the Applicable Conditions and Required Actions for LCO 3.8.1, "AC Sources—Operating."

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

Verifying the correct alignment for manual valves in the DGCW subsystem flow paths 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. 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 3.7.2.2

This SR ensures that each DGCW subsystem pump will automatically start to provide required cooling to the associated DG heat exchangers when the DG starts. These starts may be performed using actual or simulated initiation signals.

Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool or whenever movement of new fuel assemblies occurs in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool, since the potential for a release of fission products exists.

ACTIONS

A.1

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

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of a fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

(continued)

A.1

3.8 - LIMITING CONDITIONS FOR OPERATION

B. Diesel Generator Cooling Water System

~~A diesel generator cooling water (DGCW) subsystem shall be OPERABLE for each required diesel generator~~ ~~with each subsystem comprised of:~~ LA.1

~~1. One OPERABLE DGCW pump, and~~

~~2. An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water to the associated diesel generator.~~

APPLICABILITY: *MODES 1, 2, and 3* LA.2

~~When the diesel generator is required to be OPERABLE.~~

ACTION: *add proposed ACTIONS Note* A.2

Required Action A Note
ACTION A

With one or more DGCW subsystems inoperable, declare the associated diesel generator inoperable ~~and take the ACTION required by Specifications 3.9.A or 3.9.B, as applicable.~~ A.3

4.8 - SURVEILLANCE REQUIREMENTS

B. Diesel Generator Cooling Water System

Each of the required DGCW subsystems shall be demonstrated OPERABLE:

SR 3.7.2.1 1. At least once per 31 days by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position. A.4 LD.1

SR 3.7.2.2 2. At least once per ~~18~~ *24* months by verifying that each pump starts automatically upon receipt of a*n* *actual or simulated* start signal ~~for the associated diesel generator~~. L.1 LA.3 C

The following DGCW subsystems shall be OPERABLE: M.1

a. Two unit DGCW subsystems; and

b. The opposite unit DGCW subsystem capable of supporting its associated diesel generator (DG).