ATTACHMENT 3

Revision C to Quad Cities Nuclear Power Station, Units 1 and 2
Proposed Improved Technical Specifications Submittal
dated March 3, 2000

Revision C to Quad Cities 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 Quad Cities Nuclear Power Station, Units 1 and 2. The original Technical Specifications amendment request (i.e., Revision 0) was submitted to the NRC by letter dated March 3, 2000, as revised in Revisions A 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.

<u>Section 3.1</u>

1 A typographical error was noted in the CTS markups for ITS 3.1.3, in that the incorrect ITS number was noted. This change affects the CTS markup for ITS 3.1.3, pages 1 of 9, 2 of 9, 3 of 9, 7 of 9, and 9 of 9.

<u>Section 3.2</u>

1. The change committed to in the ComEd response to RAI 3.2.1-2 has been made. This change affects ITS 3.2.1, Bases pages B 3.2.1-1 and B 3.2.1-2, and the ISTS Bases markup pages B 3.2-1 and B 3.2-2.

<u>Section 3.3</u>

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 16.
2. The change committed to in the ComEd response to RAI 3.3.1.1-04 has been made. This change affects ITS 3.3.1.1, page 3.3.1.1-1 and Bases page B 3.3.1.1-23, 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 3.3.1.1-07 has been made. This change affects the Justification for Deviations to ITS 3.3.1.1, JFD 11 (page 2).
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 3.3.2.1, 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 3.3.2.1-02 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 3.3.3.1-01 has been made. This change affects ITS 3.3.3.1 page 3.3.3.1-3.

9. The change committed to in the ComEd response to RAI 3.3.3.1-06 has been made. This change affects the Discussion of Changes for ITS 3.3.3.1, DOC L.5 (page 9).

10. 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).

11. The change committed to in the ComEd response to RAI 3.3.5.1-01 has been made. This change affects ITS 3.3.5.1 page 3.3.5.1-12, the CTS markup for ITS 3.3.5.1, page 15 of 17, the Discussion of Changes for ITS 3.3.5.1, DOC M.5 (page 5), and the ISTS markup page 3.3-44.

12. The change committed to in the ComEd response to RAI 3.3.5.1-09 has been made. This change affects the Justification for Deviations to ITS 3.3.5.1, JFD 11 (page 2).

13. The change committed to in the ComEd response to RAI 3.3.5.2-01 has been made. This change affects ITS 3.3.5.2 page 3.3.5.2-4, the CTS markup for ITS 3.3.5.2, page 7 of 7, the Discussion of Changes for ITS 3.3.5.2, DOC M.2 (page 3), and the ISTS markup page 3.3-51.

14. The change committed to in the ComEd response to RAI 3.3.6.1-03 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 3.3.6.1-26 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.10 (page 3), DOC LD.1 (page 7), and DOC LE.1 (page 8), the ISTS markup, pages 3.3-55, 3.3-57, 3.3-58, insert page 3.3-58, 3.3-59, 3.3-60, 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.

15. The change committed to in the ComEd response to LaSalle RAI 3.3.4.1-03 has been made for Quad Cities for consistency between the sites. This change affects ITS 3.3.1.1, Bases pages B 3.3.1.1-19 and B 3.3.1.1-20, 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).

16. 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.

17. 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.

18. 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

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(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.

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 in the ComEd response to Dresden RAI 3.3.5.1-03 has been made for Quad Cities for consistency between the sites. This change affects the CTS markup for ITS 3.3.5.1, page 6 of 17, Discussion of Changes for ITS 3.3.5.1, DOC A.13 (page 3) and DOC L.1 (pages 9 and 10), and the No Significant Hazards Consideration for ITS 3.3.5.1, NSHC L.1 (page 1).

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. 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 page B 3.3.6.1-5 and the ISTS Bases markup insert page B 3.3-155.

24. The change committed to based on discussions with the NRC has been made. An extra period was found in the ITS description of the Main Steam Line Isolation logic. This error has been corrected. This change affects the ITS 3.3.6.1 Bases page B 3.3.6.1-2.

25. The change committed to based on discussions with the NRC has 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. 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.

27. 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 5 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 3.3.1.1-30, and B 3.3.1.1-32, 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 page B 3.3.2.2-5, 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

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page B 3.3.5.2-12, ITS 3.3.6.1 Bases page B 3.3.6.1-26, ITS 3.3.6.2 Bases page B 3.3.6.2-10, ITS 3.3.7.1 Bases page B 3.3.7.1-11, ITS 3.3.7.2 Bases page B 3.3.7.2-6, 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 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 3.3.7.2 Bases markup page B 3.3.7.2-6, the ISTS 3.3.8.1 Bases markup, pages B 3.3-225 and insert page B 3.3-225, the 3.3.8.2 ISTS Bases markup, pages B 3.3-232 and insert page B 3.3-232, ITS 3.4.5 Bases page B 3.4.5-5, 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).

28. TSTF-231 has been incorporated. This change affects ITS 3.3.1.1 Bases page B 3.3.1.1-19 and the ISTS Bases markup page B 3.3-18.

29. An error in the description of the Allowable Value for the Emergency Core Cooling System Instrumentation was noted in that a Function which has an upper and lower analytic limit was not identified in a listing. This error has been corrected. This change affects the ITS 3.3.5.1 Bases page B 3.3.5.1-10 and the ISTS Bases markup insert page B 3.3-108.

Section 3.4

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.1 (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 3.4.5-01 and related changes for consistency have been made. These changes affect ITS 3.4.5, pages 3.4.5-1, 3.4.5-2, and 3.4.5-3, and Bases pages B 3.4.5-2, B 3.4.5-3, B 3.4.5-4 and B 3.4.5-5, the Discussion of Changes for ITS 3.4.5, DOC A.2 (page 1), the ISTS markup, pages 3.4-13 and 3.4-15, the Justification for Deviations to ITS 3.4.5, JFD 7 (page 1), and the ISTS Bases markup, pages B 3.4-28, B 3.4-30, and B 3.4-32.

3. The change committed to in the ComEd response to RAI 3.4.8-01 has been made. These changes affect the CTS markup for ITS 3.4.8, page 1 of 2, and the Discussion of Changes for ITS 3.4.8, DOC A.2 (pages 1 and 2) and DOC M.1 (page 2).

4. 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.9 (pages 2 and 3).

5. 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.3 (page 2).

6. 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 page 3.4-22, and the Justification for Deviations to ITS 3.4.8, JFD 5 (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 (pages 4 and 5), 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 3.6.4.1-2 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, B 3.6.4.3-5, and B 3.6.4.3-6, the ISTS 3.6.4.1 Bases markup, pages 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 in 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 3.7.2-4, 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 7 (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. This change involves the changes proposed in a ComEd letter to the NRC (SVP-00-146) dated August 30, 2000, Request for an Amendment to the Technical Specifications For Emergency Diesel Generator Cooling Water Pump Allowed Outage Time. 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 ITS 3.7.2, pages 3.7.2-1 and 3.7.2-2, and Bases pages B 3.7.2-2, B 3.7.2-3 and B 3.7.2-4, the CTS markup for ITS 3.7.2, page 1 of 1, the Discussion of Changes for ITS 3.7.2, DOC M.1 (page 2), the ISTS markup, pages 3.7-7, insert page 3.7-7, and 3.7-8, the Justification for Deviations to ITS 3.7.2, JFD 6 (page 2), and the ISTS Bases markup, pages B 3.7-14, B 3.7-15, insert page B 3.7-15, B 3.7-16, and insert page B 3.7-16.

3. 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 (page 1).

Section 3.8

1. The change committed to in the ComEd response to RAI 3.8.1-03 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 DOC LA.3 (page 7), and the ISTS markup page 3.8-15.

2. The change committed to in the ComEd response to RAI 3.8.1-09 has been made. 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.

3. The change committed to in the ComEd response to RAI 3.8.1-11 has been made. This changes affects ITS 3.8.1 Bases page B 3.8.1-5.

4. The change committed to in the ComEd response to RAI 3.8.1-16 has been made. This change affects ITS 3.8.1 Bases page 3.8.1-22 and the ISTS Bases markup page B 3.8-19.

5. The change committed to in the ComEd response to RAI 3.8.1-19 has been made. This change affects ITS 3.8.1 Bases page 3.8.1-29 and the ISTS Bases markup insert page B 3.8-28.

6. The change committed to in the ComEd response to RAI 3.8.2-04 has been made. This change affects ITS 3.8.2 Bases page B 3.8.2-5 and the ISTS Bases markup insert page B 3.8-38.

7. The change committed to in the ComEd response to RAI 3.8.3-02 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).

8. The change committed to in the ComEd response to RAI 3.8.3-04 has been made. This change affects ITS 3.8.3 Bases page 3.8.3-1 and the ISTS Bases markup page B 3.8-41.

9. The change committed to in the ComEd response to RAI 3.8.4-03 has been made. This change affects ITS 3.8.4, page 3.8.4-6 and Bases pages B 3.8.4-3 and B 3.8.4-14, the ISTS markup, page 3.8-27 and Bases pages B 3.8-51 and B 3.8-58.

10. The change committed to in the ComEd response to RAI 3.8.4-05 has been made. This change affects ITS 3.8.4 Bases page 3.8.4-3 and the ISTS Bases markup page B 3.8-51.

11. The change committed to in the ComEd response to RAI 3.8.4-06 has been made. This change affects ITS 3.8.4 Bases page B 3.8.4-4 and the ISTS Bases markup page B 3.8-52.

12. The change committed to in the ComEd response to RAI 3.8.4-11 has been made. This change affects ITS 3.8.4, Bases pages B 3.8.4-1, B 3.8.4-2, B 3.8.4-6, B 3.8.4-7, and B 3.8.4-9, the Discussion of Changes for ITS 3.8.4, DOC LA.2 (page 3), and the ITS 3.8.4 Bases markup, insert page B 3.8.50, insert page B 3.8-53a, insert page B 3.8-53b, and insert page B 3.8-53c.

13. The change committed to in the ComEd response to RAI 3.8.6-01 has been made. This change affects ITS 3.8.6, page 3.8.6-4 and Bases page B 3.8.6-5, the ITS markup page 3.8-33, the Justification for Deviations for ITS 3.8.6, JFD 5 (page 1), and the ISTS Bases markup for ITS 3.8.6, page B 3.8-67.

14. The change committed to in the ComEd response to RAI 3.8.7-01 has been made. This change affects ITS 3.8.7, Bases page B 3.8.7-4 and the ISTS Bases markup page 3.8-81, and the Justification for Deviations to ISTS Bases 3.8.7, JFD 10 (page 1).

15. The change committed to in the ComEd response to RAI 3.8.7-02 has been made. This change affects the ITS 3.8.7, Bases pages B 3.8.7-2, B 3.8.7-3, B 3.8.7-4, B 3.8.7-11, and B 3.8.7-12, 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.

16. The change committed to in the ComEd response to RAI 3.8.8-01 has been made. This change affects the CTS markup for ITS 3.8.8, page 2 of 2, 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).

17. The change committed to in the ComEd response to Dresden RAI 3.8.1-1 has been made for Quad Cities for consistency between the sites. 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.

18. The change committed to in the ComEd response to LaSalle RAI 3.8.1-19 has been made for Quad Cities 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 18 (page 7), and the ISTS Bases markup page B 3.8-25.

19. The change committed to in the ComEd response to Dresden RAI 3.8.2-3 has been made for Quad Cities for consistency between the sites. 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 (page 5), and the No Significant Hazards Consideration for ITS 3.8.2, NSHC L.3 (page 4).

20. The change committed to in the ComEd response to Dresden RAI 3.8.5-2 has been made for Quad Cities for consistency. 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).

21. The change committed to in the ComEd response to Dresden RAI 3.8.7-1 has been made for Quad Cities for consistency between the sites. 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 from page 1, 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.

22. The change committed to in the ComEd response to Dresden RAI 3.8.7-3 has been made for Quad Cities for consistency between the sites. 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.

23. 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 pages B 3.8.1-31 and B 3.8.1-32, the CTS markup for ITS 3.8.1, page 8 of 9, the Discussion of Changes for ITS 3.8.1, DOC L.14 (page 18), the ISTS markup page 3.8-15, the Justification for Deviations to ITS 3.8.1, JFD 14 (pages 5 and 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).

24. 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.

25. 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.

26. 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.

27. 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 3.8.4-13 and the ISTS Bases markup page B 3.8-57.

28. 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).

29. 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 page B 3.8.7-1 and the ISTS Bases markup insert page B 3.8-79.

30. 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.

31. 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 3.8.7-4 and the ISTS Bases markup page B 3.8-81.

32. 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 electrical distribution systems 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.

33. This change involves the changes proposed in a ComEd letter to the NRC (SVP-00-145) dated August 30, 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, but a new DOC A.13 was added per another change) and DOC M.5 (pages 5 and 6).

34. 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.

35. The term "Division 1 or 2" has been added to ITS 3.8.4 Required Actions B.2, C.2, and D.1 for consistency with the associated Conditions. In addition, a typographical error was corrected in Condition F (the word "Associated" should not be capitalized) and in the ACTIONS D.1 and D.2 Bases (the time was changed from "7 days" to "72 hours"). These changes affect ITS 3.8.4, pages 3.8.4-2 and 3.8.4-3, and Bases page B 3.8.4-8, the ISTS markup pages 3.8-24, insert page 3.8-24a, and insert page 3.8-24b, the Justification for Deviations to ITS 3.8.4, JFD 6 (page 1), and the ISTS Bases markup insert page B 3.8-53c.

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, addition of word "of" to Applicable Safety Analyses Bases section, and addition of "250" to the SR 3.8.5.1 Bases). 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.

37. A typographical error has been corrected in the ITS 3.8.7 Bases (the proper bus number has been provided). This change affects ITS 3.8.7 Bases page B 3.8.7-1 and the ISTS Bases markup page B 3.8-79.

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 4 (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 the 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 17 and 11 of 17, 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 6, 4 of 6, and 5 of 6, the Discussion of Changes for ITS 5.6, DOC LA.2 (page 3), and the ISTS markup pages 5.0-20, insert page 5.0-20a, and insert page 5.0-20b.

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.

Quad Cities ITS Rev. C Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 1	
CHAPTERS 1.0 AND 2.0 AND SECTION 3.0	
DISCARD	**INSERT**
ITS pages 1.1-1 and 1.1-2	ITS pages 1.1-1 and 1.1-2
ITS page 1.1-4	ITS page 1.1-4
CTS markup for Chapter 1.0 page 1 of 12	CTS markup for Chapter 1.0 page 1 of 12
CTS markup for Chapter 1.0 page 3 of 12	CTS markup for Chapter 1.0 page 3 of 12
CTS markup for Chapter 1.0 page 5 of 12	CTS markup for Chapter 1.0 page 5 of 12
ISTS markup pages 1.1-1 and 1.1-2	ISTS markup pages 1.1-1 and 1.1-2
ISTS markup page 1.1-5	ISTS markup page 1.1-5

VOLUME 2	
SECTIONS 3.1 AND 3.2	
DISCARD	**INSERT**
CTS markup for Specification 3.1.3 pages 1 of 9 through 3 of 9	CTS markup for Specification 3.1.3 pages 1 of 9 through 3 of 9
CTS markup for Specification 3.1.3 page 7 of 9	CTS markup for Specification 3.1.3 page 7 of 9
CTS markup for Specification 3.1.3 page 9 of 9	CTS markup for Specification 3.1.3 page 9 of 9
ITS Bases pages B 3.2.1-1 and B 3.2.1-2	ITS Bases pages B 3.2.1-1 and B 3.2.1-2
ISTS Bases markup pages B 3.2-1 and B 3.2-2	ISTS Bases markup pages B 3.2-1 and B 3.2-2

Quad Cities ITS Rev. C Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 3	
SECTION 3.3	
DISCARD	**INSERT**
ITS page 3.3.1.1-1	ITS page 3.3.1.1-1
ITS page 3.3.3.1-3	ITS page 3.3.3.1-3
ITS page 3.3.5.1-12	ITS page 3.3.5.1-12
ITS page 3.3.5.2-4	ITS page 3.3.5.2-4
ITS pages 3.3.6.1-4 through 3.3.6.1-7	ITS pages 3.3.6.1-4 through 3.3.6.1-7
ITS Bases pages B 3.3.1.1-19 and B 3.3.1.1-20	ITS Bases pages B 3.3.1.1-19 and B 3.3.1.1-20
ITS Bases pages B 3.3.1.1-23 and B 3.3.1.1-24	ITS Bases pages B 3.3.1.1-23 and B 3.3.1.1-24
ITS Bases pages B 3.3.1.1-29 through B 3.3.1.1-36	ITS Bases pages B 3.3.1.1-29 through B 3.3.1.1-36
ITS Bases page B 3.3.1.2-2	ITS Bases page B 3.3.1.2-2
ITS Bases pages B 3.3.1.2-8 through B 3.3.1.2-10	ITS Bases pages B 3.3.1.2-8 through B 3.3.1.2-10
ITS Bases pages B 3.3.2.1-9 through B 3.3.2.1-13	ITS Bases pages B 3.3.2.1-9 through B 3.3.2.1-14
ITS Bases pages B 3.3.2.2-5 and B 3.3.2.2-6	ITS Bases pages B 3.3.2.2-5 and B 3.3.2.2-6
ITS Bases page B 3.3.3.1-5	ITS Bases page B 3.3.3.1-5
ITS Bases pages B 3.3.4.1-9 and B 3.3.4.1-10	ITS Bases pages B 3.3.4.1-9 and B 3.3.4.1-10
ITS Bases pages B 3.3.5.1-4 through B 3.3.5.1-7	ITS Bases pages B 3.3.5.1-4 through B 3.3.5.1-7
ITS Bases page B 3.3.5.1-10	ITS Bases page B 3.3.5.1-10
ITS Bases pages B 3.3.5.1-41 and B 3.3.5.1-42	ITS Bases pages B 3.3.5.1-41 through B 3.3.5.1-43
ITS Bases pages B 3.3.5.2-12 and B 3.3.5.2-13	ITS Bases pages B 3.3.5.2-12 and B 3.3.5.2-13
ITS Bases page B 3.3.6.1-2	ITS Bases page B 3.3.6.1-2
ITS Bases pages B 3.3.6.1-5 and B 3.3.6.1-6	ITS Bases pages B 3.3.6.1-5 and B 3.3.6.1-6
ITS Bases page B 3.3.6.1-9	ITS Bases page B 3.3.6.1-9
ITS Bases pages B 3.3.6.1-26 through B 3.3.6.1-28	ITS Bases pages B 3.3.6.1-26 through B 3.3.6.1-28
ITS Bases pages B 3.3.6.2-10 through B 3.3.6.2-12	ITS Bases pages B 3.3.6.2-10 through B 3.3.6.2-12
ITS Bases pages B 3.3.7.1-11 through B 3.3.7.1-13	ITS Bases pages B 3.3.7.1-11 through B 3.3.7.1-13
ITS Bases pages B 3.3.7.2-6 and B 3.3.7.2-7	ITS Bases pages B 3.3.7.2-6 and B 3.3.7.2-7

Quad Cities ITS Rev. C Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 3	
SECTION 3.3	
DISCARD	**INSERT**
ITS Bases pages B 3.3.8.1-7 and B 3.3.8.1-8	ITS Bases pages B 3.3.8.1-7 and B 3.3.8.1-8
ITS Bases page B 3.3.8.2-2	ITS Bases page B 3.3.8.2-2
ITS Bases pages B 3.3.8.2-6 and B 3.3.8.2-7	ITS Bases pages B 3.3.8.2-6 and B 3.3.8.2-7
CTS markup for Specification 3.3.1.1 page 12 of 16	CTS markup for Specification 3.3.1.1 page 12 of 16
CTS markup for Specification 3.3.1.2 page 2 of 3	CTS markup for Specification 3.3.1.2 page 2 of 3
Discussion of Changes for ITS 3.3.1.2 page 2	Discussion of Changes for ITS 3.3.1.2 page 2
Discussion of Changes for ITS 3.3.2.1 pages 3 through 7	Discussion of Changes for ITS 3.3.2.1 pages 3 through 7
Discussion of Changes for ITS 3.3.3.1 page 9	Discussion of Changes for ITS 3.3.3.1 page 9
Discussion of Changes for ITS 3.3.4.1 pages 2 through 9	Discussion of Changes for ITS 3.3.4.1 pages 2 through 10
CTS markup for Specification 3.3.5.1 page 6 of 17	CTS markup for Specification 3.3.5.1 page 6 of 17
CTS Markup for Specification 3.3.5.1 page 15 of 17	CTS Markup for Specification 3.3.5.1 page 15 of 17
Discussion of Changes for ITS 3.3.5.1 pages 3 through 5	Discussion of Changes for ITS 3.3.5.1 pages 3 through 5
Discussion of Changes for ITS 3.3.5.1 page 9	Discussion of Changes for ITS 3.3.5.1 pages 9 and 10
CTS Markup for Specification 3.3.5.2 page 7 of 7	CTS Markup for Specification 3.3.5.2 page 7 of 7
Discussion of Changes for ITS 3.3.5.2 pages 1 through 6	Discussion of Changes for ITS 3.3.5.2 pages 1 through 6
CTS Markup for Specification 3.3.6.1 page 1 of 12	CTS Markup for Specification 3.3.6.1 page 1 of 12
CTS Markup for Specification 3.3.6.1 pages10 of 12 through 12 of 12	CTS Markup for Specification 3.3.6.1 pages 10 of 12 through 12 of 12
Discussion of Changes for ITS 3.3.6.1 pages 3 and 4	Discussion of Changes for ITS 3.3.6.1 pages 3 and 4
Discussion of Changes for ITS 3.3.6.1 pages 7 and 8	Discussion of Changes for ITS 3.3.6.1 pages 7 and 8

VOLUME 4	
SECTION 3.3	
DISCARD	**INSERT**
ISTS markup page 3.3-1	ISTS markup page 3.3-1
Justification for Deviations to ITS 3.3.1.1 page 2	Justification for Deviations to ITS 3.3.1.1 page 2
Justification for Deviations to ITS 3.3.2.2 page 1	Justification for Deviations to ITS 3.3.2.2 page 1
ISTS markup page 3.3-44	ISTS markup page 3.3-44
Justification for Deviations to ITS 3.3.5.1 page 2	Justification for Deviations to ITS 3.3.5.1 page 2
ISTS markup page 3.3-51	ISTS markup page 3.3-51
ISTS markup page 3.3-55	ISTS markup page 3.3-55
ISTS markup page 3.3-57	ISTS markup page 3.3-57
ISTS markup page 3.3-58 and insert page 3.3-58	ISTS markup page 3.3-58 and insert page 3.3-58
ISTS markup page 3.3-59	ISTS markup page 3.3-59
ISTS markup page 3.3-60 and insert page 3.3-60	ISTS markup page 3.3-60 and insert page 3.3-60
ISTS markup pages 3.3-61 and 3.3-62	ISTS markup pages 3.3-61 and 3.3-62
Justification for Deviations to ITS 3.3.6.1 pages 1 and 2	Justification for Deviations to ITS 3.3.6.1 pages 1 and 2
ISTS Bases markup pages B 3.3-18 and B 3.3-19	ISTS Bases markup page B 3.3-18 and B 3.3-19
ISTS Bases markup insert page B 3.3-21	ISTS Bases markup insert page B 3.3-21
ISTS Bases markup page B 3.3-27 and insert page B 3.3-27	ISTS Bases markup page B 3.3-27 and insert page B 3.3-27
ISTS Bases markup page B 3.3-29	ISTS Bases markup page B 3.3-29 and Bases markup insert page B 3.3-29
Justification for Deviations to ITS Bases 3.3.1.1 page 1	Justification for Deviations to ITS Bases 3.3.1.1 page 1
ISTS Bases markup page B 3.3-36	ISTS Bases markup page B 3.3-36
ISTS Bases markup page B 3.3-41	ISTS Bases markup page B 3.3-41 and insert page B 3.3-41
ISTS Bases markup page B 3.3-51	ISTS Bases markup page B 3.3-51 and insert page B 3.3-51
ISTS Bases markup page B 3.3-52	ISTS Bases markup page B 3.3-52
ISTS Bases markup page B 3.3-53	ISTS Bases markup page B 3.3-53 and insert page B 3.3-53
ISTS Bases markup insert page B 3.3-54	ISTS Bases markup insert page B 3.3-54
ISTS Bases markup page B 3.3-61	ISTS Bases markup page B 3.3-61 and insert page B 3.3-61
ISTS Bases markup page B 3.3-66	ISTS Bases markup page B 3.3-66

Quad Cities ITS Rev. C Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 4	
SECTION 3.3	
DISCARD	**INSERT**
ISTS Bases markup page B 3.3-87	ISTS Bases markup page B 3.3-87
ISTS Bases markup page B 3.3-98	ISTS Bases markup page B 3.3-98 and insert page B 3.3-98
ISTS Bases markup insert page B 3.3-103	ISTS Bases markup insert page B 3.3-103
ISTS Bases markup insert page B 3.3-108	ISTS Bases markup insert page B 3.3-108
ISTS Bases markup page B 3.3-136	ISTS Bases markup page B 3.3-136 and insert page B 3.3-136
ISTS Bases markup page B 3.3-149	ISTS Bases markup page B 3.3-149 and insert page B 3.3-149
ISTS Bases markup insert page B 3.3-155	ISTS Bases markup insert page B 3.3-155
ISTS Bases markup insert page B 3.3-159	ISTS Bases markup insert page B 3.3-159
ISTS Bases markup page B 3.3-181	ISTS Bases markup page B 3.3-181 and insert page B 3.3-181
ISTS Bases markup pages B 3.3-182 and B 3.3-183	ISTS Bases markup pages B 3.3-182 and B 3.3-183
ISTS Bases markup page B 3.3-194	ISTS Bases markup page B 3.3-194 and insert page B 3.3-194
ISTS Bases markup page B 3.3-205	ISTS Bases markup page B 3.3-205
ISTS Bases markup page B 3.3-217	ISTS Bases markup page B 3.3-217 and insert page B 3.3-217
ISTS Bases markup insert page B 3.3.7.2-6	ISTS Bases markup insert page B 3.3.7.2-6
ISTS Bases markup insert page B 3.3.7.2-7	ISTS Bases markup insert page B 3.3.7.2-7
ISTS Bases markup page B 3.3-225	ISTS Bases markup page B 3.3-225 and insert page B 3.3-225
ISTS Bases markup page B 3.3-228	ISTS Bases markup page B 3.3-228
ISTS Bases markup page B 3.3-232	ISTS Bases markup page B 3.3-232 and insert page B 3.3-232
No Significant Hazards Consideration for ITS 3.3.5.1 page 1	No Significant Hazards Consideration for ITS 3.3.5.1 page 1

Quad Cities ITS Rev. C Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 5	
SECTIONS 3.4 AND 3.5	
DISCARD	**INSERT**
ITS pages 3.4.5-1 through 3.4.5-3	ITS pages 3.4.5-1 through 3.4.5-3
ITS page 3.4.8-2	ITS page 3.4.8-2
ITS Bases pages B 3.4.5-2 through B 3.4.5-6	ITS Bases pages B 3.4.5-2 through B 3.4.5-6
CTS markup for Specification 3.4.1 page 3 of 3	CTS markup for Specification 3.4.1 page 3 of 3
Discussion of Changes for ITS 3.4.1 pages 2 through 5	Discussion of Changes for ITS 3.4.1 pages 2 through 5
CTS markup for Specification 3.4.3 page 3 of 4	CTS markup for Specification 3.4.3 page 3 of 4
Discussion of Changes for ITS 3.4.3 page 2	Discussion of Changes for ITS 3.4.3 page 2
Discussion of Changes for ITS 3.4.5 page 1	Discussion of Changes for ITS 3.4.5 page 1
CTS markup for Specification 3.4.8 page 1 of 2	CTS markup for Specification 3.4.8 page 1 of 2
Discussion of Changes for ITS 3.4.8 pages 1 through 4	Discussion of Changes for ITS 3.4.8 pages 1 through 4
CTS markup for Specification 3.4.9 page 2 of 6	CTS markup for Specification 3.4.9 page 2 of 6
Discussion of Changes for ITS 3.4.9 pages 2 through 5	Discussion of Changes for ITS 3.4.9 pages 2 through 5
ISTS markup page 3.4-13	ISTS markup page 3.4-13
ISTS markup page 3.4-15	ISTS markup page 3.4-15
Justification for Deviations to ITS 3.4.5 page 1	Justification for Deviations to ITS 3.4.5 page 1
ISTS markup page 3.4-22	ISTS markup page 3.4-22
Justification for Deviations to ITS 3.4.8 page 1	Justification for Deviations to ITS 3.4.8 page 1
ISTS Bases markup page B 3.4-28	ISTS Bases markup page B 3.4-28
ISTS Bases markup page B 3.4-30	ISTS Bases markup page B 3.4-30
ISTS Bases markup page B 3.4-32 and insert page B 3.4-32	ISTS Bases markup page B 3.4-32 and insert page B 3.4-32
N/A	No Significant Hazards Consideration for ITS 3.4.1 page 4

Quad Cities ITS Rev. C Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 6	
SECTION 3.6	
DISCARD	**INSERT**
ITS page 3.6.1.1-2	ITS page 3.6.1.1-2
ITS Bases page B 3.6.1.1-5	ITS Bases page B 3.6.1.1-5
ITS Bases pages B 3.6.1.7-2 through B 3.6.1.7-6	ITS Bases pages B 3.6.1.7-2 through B 3.6.1.7-6
ITS Bases pages B 3.6.4.1-3 and B 3.6.4.1-4	ITS Bases pages B 3.6.4.1-3 and B 3.6.4.1-4
ITS Bases pages B 3.6.4.2-5 through B 3.6.4.2-7	ITS Bases pages B 3.6.4.2-5 through B 3.6.4.2-7
ITS Bases pages B 3.6.4.3-4 through B 3.6.4.3-7	ITS Bases pages B 3.6.4.3-4 through B 3.6.4.3-7
CTS markup for Specification 3.6.1.1 page 3 of 3	CTS markup for Specification 3.6.1.1 page 3 of 3
Discussion of Changes for ITS 3.6.1.1 pages 2 through 5	Discussion of Changes for ITS 3.6.1.1 pages 2 through 5
CTS markup for Specification 3.6.4.2 page 3 of 3	CTS markup for Specification 3.6.4.2 page 3 of 3
Discussion of Changes for ITS 3.6.4.2 pages 2 through 5	Discussion of Changes for ITS 3.6.4.2 pages 2 through 6

VOLUME 7	
SECTION 3.6	
DISCARD	**INSERT**
ISTS markup page 3.6-2	ISTS markup page 3.6-2
ISTS Bases markup insert page B 3.6-4	ISTS Bases markup insert page B 3.6-4
ISTS Bases markup page B 3.6-5	ISTS Bases markup page B 3.6-5
ISTS Bases markup page B 3.6-43	ISTS Bases markup page B 3.6-43
ISTS Bases markup page B 3.6-44	ISTS Bases markup page B 3.6-44
Justification for Deviations to ITS Bases 3.6.1.7 page 1	Justification for Deviations to ITS Bases 3.6.1.7 page 1
ISTS Bases markup page B 3.6-99	ISTS Bases markup page B 3.6-99
ISTS Bases markup insert page B 3.6-99	None
ISTS Bases markup page B 3.6-100	ISTS Bases markup page B 3.6-100
ISTS Bases markup page B 3.6-106	ISTS Bases markup page B 3.6-106
ISTS Bases markup insert page B 3.6-106	None
ISTS Bases markup page B 3.6-112	ISTS Bases markup page B 3.6-112
ISTS Bases markup insert page B 3.6-112	ISTS Bases markup insert page B 3.6-112
ISTS Bases markup page B 3.6-113	ISTS Bases markup page B 3.6-113
ISTS Bases markup insert page B 3.6-113	None
No Significant Hazards Consideration for ITS 3.6.1.1 pages 5 and 6	N/A

VOLUME 8	
SECTION 3.7	
DISCARD	**INSERT**
ITS pages 3.7.2-1 and 3.7.2-2	ITS pages 3.7.2-1 and 3.7.2-2
ITS Bases pages B 3.7.2-2 through B 3.7.2-4	ITS Bases pages B 3.7.2-2 through B 3.7.2-4
ITS Bases page B 3.7.8-2	ITS Bases page B 3.7.8-2
CTS markup for Specification 3.7.2 page 1 of 1	CTS markup for Specification 3.7.2 page 1 of 1
Discussion of Changes for ITS 3.7.2 pages 2 through 4	Discussion of Changes for ITS 3.7.2 pages 2 through 4
ISTS markup page 3.7-7 and insert page 3.3-7	ISTS markup page 3.7-7 and insert page 3.3-7
ISTS markup page 3.7-8	ISTS markup page 3.7-8
Justification for Deviations to ITS 3.7.2 page 2	Justification for Deviations to ITS 3.7.2 page 2
ISTS Bases markup page B 3.7-14	ISTS Bases markup page B 3.7-14
ISTS Bases markup page B 3.7-15 and insert page B 3.7-15	ISTS Bases markup page B 3.7-15 and insert page B 3.7-15
ISTS Bases markup page B 3.7-16	ISTS Bases markup page B 3.7-16 and insert page B 3.7-16
ISTS Bases markup page B 3.7-38 and insert page B 3.7-38	ISTS Bases markup page B 3.7-38
Justification for Deviations to ITS Bases 3.7.8 page 1	Justification for Deviations to ITS Bases 3.7.8 page 1
No Significant Hazards Consideration for ITS 3.7.2 page 1	No Significant Hazards Consideration for ITS 3.7.2 page 1

VOLUME 9	
SECTION 3.8	
DISCARD	**INSERT**
ITS page 3.8.1-4	ITS page 3.8.1-4
ITS page 3.8.1-6	ITS page 3.8.1-6
ITS page 3.8.1-11	ITS page 3.8.1-11
ITS page 3.8.1-14	ITS page 3.8.1-14
ITS pages 3.8.4-2 and 3.8.4-3	ITS pages 3.8.4-2 and 3.8.4-3
ITS page 3.8.4-6	ITS page 3.8.4-6
ITS pages 3.8.5-1 and 3.8.5-2	ITS pages 3.8.5-1 and 3.8.5-2
ITS page 3.8.6-4	ITS page 3.8.6-4
ITS pages 3.8.7-1 through 3.8.7-3	ITS pages 3.8.7-1 through 3.8.7-3
ITS Bases pages B 3.8.1-5 through B 3.8.1-11	ITS Bases pages B 3.8.1-5 through B 3.8.1-11
ITS Bases page B 3.8.1-15	ITS Bases page B 3.8.1-15
ITS Bases page B 3.8.1-19	ITS Bases page B 3.8.1-19
ITS Bases pages B 3.8.1-21 through B 3.8.1-23	ITS Bases pages B 3.8.1-21 through B 3.8.1-23
ITS Bases pages B 3.8.1-27 through B 3.8.1-35	ITS Bases pages B 3.8.1-27 through B 3.8.1-34
ITS Bases pages B 3.8.2-5 through B 3.8.2-7	ITS Bases pages B 3.8.2-5 through B 3.8.2-7
ITS Bases page B 3.8.3-1	ITS Bases page B 3.8.3-1
ITS Bases pages B 3.8.4-1 through B 3.8.4-14	ITS Bases pages B 3.8.4-1 through B 3.8.4-15
ITS Bases pages B 3.8.5-1 through B 3.8.5-4	ITS Bases pages B 3.8.5-1 through B 3.8.5-5
ITS Bases page B 3.8.6-5	ITS Bases page B 3.8.6-5
ITS Bases pages B 3.8.7-1 through B 3.8.7-14	ITS Bases pages B 3.8.7-1 through B 3.8.7-12
ITS Bases page B 3.8.8-2	ITS Bases page B 3.8.8-2
CTS markup for Specification 3.8.1 page 2 of 9	CTS markup for Specification 3.8.1 page 2 of 9
CTS markup for Specification 3.8.1 pages 4 of 9 through 6 of 9	CTS markup for Specification 3.8.1 pages 4 of 9 through 6 of 9
CTS markup for Specification 3.8.1 page 8 of 9	CTS markup for Specification 3.8.1 page 8 of 9
Discussion of Changes for ITS 3.8.1 pages 3 through 18	Discussion of Changes for ITS 3.8.1 pages 3 through 19
CTS markup for Specification 3.8.2 page 1 of 2	CTS markup for Specification 3.8.2 page 1 of 2
Discussion of Changes for ITS 3.8.2 page 5	Discussion of Changes for ITS 3.8.2 page 5

VOLUME 9	
SECTION 3.8	
DISCARD	**INSERT**
CTS markup for Specification 3.8.3 page 5 of 6	CTS markup for Specification 3.8.3 page 5 of 6
Discussion of Changes for ITS 3.8.3 pages 2 and 3	Discussion of Changes for ITS 3.8.3 pages 2 and 3
Discussion of Changes for ITS 3.8.4 page 3	Discussion of Changes for ITS 3.8.4 page 3
CTS markup for Specification 3.8.5 page 1 of 1	CTS markup for Specification 3.8.5 page 1 of 1
Discussion of Changes for ITS 3.8.5 pages 1 through 3	Discussion of Changes for ITS 3.8.5 pages 1 through 3
CTS markup for Specification 3.8.7 pages 1 of 2 and 2 of 2	CTS markup for Specification 3.8.7 pages 1 of 2 and 2 of 2
Discussion of Changes for ITS 3.8.7 pages 1 through 4	Discussion of Changes for ITS 3.8.7 pages 1 through 4
CTS markup for Specification 3.8.8 page 2 of 2	CTS markup for Specification 3.8.8 page 2 of 2
Discussion of Changes for ITS 3.8.8 page 3	Discussion of Changes for ITS 3.8.8 page 3
ISTS markup page 3.8-4	ISTS markup page 3.8-4
ISTS markup page 3.8-6	ISTS markup page 3.8-6
ISTS markup page 3.8-11	ISTS markup page 3.8-11
ISTS markup page 3.8-15	ISTS markup page 3.8-15
Justification for Deviations to ITS 3.8.1 pages 5 and 6	Justification for Deviations to ITS 3.8.1 pages 5 through 7
ISTS markup page 3.8-24 and insert pages 3.8-24a and 3.8-24b	ISTS markup page 3.8-24 and insert pages 3.8-24a and 3.8-24b
Justification for Deviations to ITS 3.8.4 page 1	Justification for Deviations to ITS 3.8.4 page 1
ISTS markup page 3.8-27	ISTS markup page 3.8-27
ISTS markup pages 3.8-28 and 3.8-29	ISTS markup pages 3.8-28 and 3.8-29
Justification for Deviations to ITS 3.8.5 page 1	Justification for Deviations to ITS 3.8.5 page 1
ISTS markup page 3.8-33	ISTS markup page 3.8-33
Justification for Deviations to ITS 3.8.6 page 1	Justification for Deviations to ITS 3.8.6 page 1
ISTS markup page 3.8-36	ISTS markup page 3.8-36
ISTS markup page 3.8-38 and insert page 3.8-38	ISTS markup page 3.8-38 and insert page 3.8-38
ISTS markup page 3.8-39	ISTS markup page 3.8-39
Justification for Deviations to ITS 3.8.7 pages 1 and 2	Justification for Deviations to ITS 3.8.7 pages 1 and 2

VOLUME 9	
SECTION 3.8	
DISCARD	**INSERT**
ISTS Bases markup page B 3.8-4 and insert page B 3.8-4	ISTS Bases markup page B 3.8-4 and insert page B 3.8-4
ISTS Bases markup page B 3.8-13	ISTS Bases markup page B 3.8-13
ISTS Bases markup insert page B 3.8-17	ISTS Bases markup insert page B 3.8-17
ISTS Bases markup insert page B 3.8-18	ISTS Bases markup insert page B 3.8-18
ISTS Bases markup pages B 3.8-19 and B 3.8-20	ISTS Bases markup pages B 3.8-19 and B 3.8-20
ISTS Bases markup page B 3.8-25	ISTS Bases markup page B 3.8-25
ISTS Bases markup insert page B 3.8-28	ISTS Bases markup insert page B 3.8-28
ISTS Bases markup page B 3.8-30	ISTS Bases markup page B 3.8-30 and insert page B 3.8-30
ISTS Bases markup page B 3.8-38	ISTS Bases markup page B 3.8-38
ISTS Bases markup page B 3.8-41	ISTS Bases markup page B 3.8-41
ISTS Bases markup insert page B 3.8-50	ISTS Bases markup insert page B 3.8-50
ISTS Bases markup pages B 3.8-51 and B 3.8-52	ISTS Bases markup pages B 3.8-51 and B 3.8-52
ISTS Bases markup insert pages B 3.8-53a through B 3.8-53c	ISTS Bases markup insert pages B 3.8-53a through B 3.8-53d
ISTS Bases markup page B 3.8-57	ISTS Bases markup page B 3.8-57
ISTS Bases markup page B 3.8-58	ISTS Bases markup page B 3.8-58
ISTS Bases markup page B 3.8-60	ISTS Bases markup page B 3.8-60 and insert page B 3.8-60
ISTS Bases markup page B 3.8-61 and insert page B 3.8-61	ISTS Bases markup page B 3.8-61 and insert page B 3.8-61
ISTS Bases markup page B 3.8-62	ISTS Bases markup page B 3.8-62
ISTS Bases markup page B 3.8-67	ISTS Bases markup page B 3.8-67
Justification for Deviations to ITSBases 3.8.6 page 1	Justification for Deviations to ITS Bases 3.8.6 page 1
ISTS Bases markup pages B 3.8-75 through B 3.8-77	ISTS Bases markup pages B 3.8-75 through B 3.8-77
ISTS Bases markup page B 3.8-79 and insert page B 3.8-79	ISTS Bases markup page B 3.8-79 and insert page B 3.8-79
ISTS Bases markup page B 3.8-80	ISTS Bases markup page B 3.8-80
ISTS Bases markup page B 3.8-81 and insert page B 3.8-81	ISTS Bases markup page B 3.8-81 and insert page B 3.8-81
ISTS Bases markup pages B 3.8-82 and B 3.8-83	ISTS Bases markup pages B 3.8-82 and B 3.8-83
ISTS Bases markup page B 3.8-84 and insert pages B 3.8-84a and B 3.8-84b	ISTS Bases markup page B 3.8-84

Quad Cities ITS Rev. C Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 9	
SECTION 3.8	
DISCARD	**INSERT**
ISTS Bases markup page B 3.8-85	ISTS Bases markup page B 3.8-85
ISTS Bases markup page B 3.8-86 and insert page B 3.8-86	ISTS Bases markup page B 3.8-86 and insert page B 3.8-86
ISTS Bases markup page B 3.8-87	ISTS Bases markup page B 3.8-87
ISTS Bases markup page B 3.8-88 and insert page B 3.8-88	ISTS Bases markup page B 3.8-88 and insert pages B 3.8-88a and B 3.8-88b
Justification for Deviations to ITS Bases 3.8.7 page 1	Justification for Deviations to ITS Bases 3.8.7 page 1
ISTS Bases markup page B 3.8-90	ISTS Bases markup page B 3.8-90
N/A	No Significant Hazards Consideration for ITS 3.8.1 page 18
N/A	No Significant Hazards Consideration for ITS 3.8.2 page 4
No Significant Hazards Consideration for ITS 3.8.3 page 3	N/A
N/A	No Significant Hazards Consideration for ITS 3.8.5 page 2
No Significant Hazards Consideration for ITS 3.8.8 page 1	No Significant Hazards Consideration for ITS 3.8.8 page 1

Quad Cities ITS Rev. C Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 10	
SECTIONS 3.9 AND 3.10	
DISCARD	**INSERT**
ITS page 3.9.1-1	ITS page 3.9.1-1
ITS Bases pages B 3.9.1-4 and B 3.9.1-5	ITS Bases pages B 3.9.1-4 and B 3.9.1-5
ITS Bases pages B 3.9.2-3 and B 3.9.2-4	ITS Bases pages B 3.9.2-3 and B 3.9.2-4
Discussion of Changes for ITS 3.9.1 page 4	Discussion of Changes for ITS 3.9.1 page 4
ISTS markup page 3.9-1	ISTS markup page 3.9-1
ISTS markup insert page 3.9-1	ISTS markup insert page 3.9-1
Justification for Deviations to ITS 3.9.1 page 1	Justification for Deviations to ITS 3.9.1 page 1
ISTS Bases markup page B 3.9-3	ISTS Bases markup page B 3.9-3
ISTS Bases markup insert page B 3.9-3	ISTS Bases markup insert page B 3.9-3
ISTS Bases markup page B 3.9-4	ISTS Bases markup page B 3.9-4 and insert page B 3.9-4
ISTS Bases markup page B 3.9-7	ISTS Bases markup page B 3.9-7 and insert page B 3.9-7
Justification for Deviations to ITS Bases 3.9.2 page 1	Justification for Deviations to ITS Bases 3.9.2 page 1
No Significant Hazards Consideration for ITS 3.9.1 pages 3 and 4	No Significant Hazards Consideration for ITS 3.9.1 pages 3 and 4

VOLUME 11	
CHAPTERS 4.0 AND 5.0	
DISCARD	**INSERT**
ITS page 5.2-2	ITS page 5.2-2
ITS page 5.5-1	ITS page 5.5-1
ITS pages 5.5-5 and 5.5-6	ITS pages 5.5-5 and 5.5-6
ITS pages 5.5-9 and 5.5-10	ITS pages 5.5-9 and 5.5-10
ITS pages 5.6-3 through 5.6-5	ITS pages 5.6-3 through 5.6-5
CTS markup for Specification 5.1 page 1 of 1	CTS markup for Specification 5.1 page 1 of 1
Discussion of Changes for ITS 5.1 pages 1 and 2	Discussion of Changes for ITS 5.1 pages 1 and 2
CTS markup for Specification 5.2 page 3 of 3	CTS markup for Specification 5.2 page 3 of 3
Discussion of Changes for ITS 5.2 pages 1 through 3	Discussion of Changes for ITS 5.2 pages 1 through 3
CTS markup for Specification 5.5 page 8 of 17	CTS markup for Specification 5.5 page 8 of 17
CTS markup for Specification 5.5 page 11 of 17	CTS markup for Specification 5.5 page 11 of 17
Discussion of Changes for ITS 5.5 page 3	Discussion of Changes for ITS 5.5 page 3
CTS markup for Specification 5.6 pages 3 of 6 through 5 of 6	CTS markup for Specification 5.6 pages 3 of 6 through 5 of 6
Discussion of Changes for ITS 5.6 page 3	Discussion of Changes for ITS 5.6 pages 3 and 4
ISTS markup pages 5.0-3 and 5.0-4	ISTS markup pages 5.0-3 and 5.0-4
Justification for Deviations to ITS 5.2 page 1	Justification for Deviations to ITS 5.2 page 1
ISTS markup insert page 5.0-11	ISTS markup insert page 5.0-11
ISTS markup page 5.0-15	ISTS markup page 5.0-15
ISTS markup page 5.0-20	ISTS markup page 5.0-20
ISTS markup insert pages 5.0-20a and 5.0-20b	ISTS markup insert pages 5.0-20a and 5.0-20b

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

(continued)

1.1 Definitions

CHANNEL CHECK (continued)	status derived from independent instrument channels measuring the same parameter.
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.
MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)	The MFLPD shall be the largest value of the fraction of limiting power density (FLPD) in the core. The FLPD shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.
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

(continued)

1.0 DEFINITIONS

Notes to Definitions

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications. of this section are and Bases A.1

ACTION 5

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

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) for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle. LHGRs at the height A.1

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.3 all devices in the channel required for channel OPERABILITY and

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the CHANNEL output such that it responds with the necessary range and accuracy to known values of the parameter which the CHANNEL monitors. The CHANNEL CALIBRATION shall encompass the entire CHANNEL including the required sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is calibrated. in that Insert 1 A.4 means of 2 A.1 C C

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of CHANNEL behavior during operation by observation. This determination shall include, where possible, comparison of the CHANNEL indication and/or status with other indications and/or status derived from independent instrument CHANNELs measuring the same parameter. 1 2 to A.1

1.1 1.0 DEFINITIONS

CHANNEL FUNCTIONAL TEST
A CHANNEL FUNCTIONAL TEST shall be: 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 trips~~. of all devices in the channel required for OPERABILITY A.3

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

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

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 ~~control~~ cell. 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) is ~~shall be~~ the unit specific document that provides cycle specific parameter ~~core operating~~ limits for the current reload ~~operating~~ cycle. These cycle specific ~~core operating~~ limits shall be determined for each ~~operating~~ reload cycle in accordance with Specification 5.6.5 ~~6.9~~. 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 that power in the assembly which is calculated by application of the applicable NRC approved critical power correlation to cause some point in the assembly to experience transition boiling, divided by the actual assembly power.~~ is that appropriate (s) operating 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) that ~~which~~ 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 two additional thyroid dose conversion factor methods A.1 L.2

1.1 **1.0 DEFINITIONS**

required for OPERABILITY A.3

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.1

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

The ~~MAXIMUM FRACTION OF LIMITING POWER DENSITY~~ (MFLPD) shall be the ~~highest~~ largest value of the fraction of limiting power density (FLPD) ~~which exists~~ in the core ~~applicable to GE fuel~~. Insert definition of FLPD from page 1-3 A.1 A.5

MINIMUM CRITICAL POWER RATIO (MCPR)

The ~~MINIMUM CRITICAL POWER RATIO~~ (MCPR) shall be the smallest critical power ratio (CPR) ~~which~~ that exists in the core for each class of fuel. Insert definition of CPR from page 1-2 A.1 A.5

~~OFFSITE DOSE CALCULATION MANUAL (ODCM)~~

~~The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specification 6.9.~~ A.9 moved to Specification 5.5

OPERABLE - OPERABILITY

A system, subsystem, ~~train~~ 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 ~~or~~ and seal water, lubrication, ~~or~~ other auxiliary equipment that are required for the system, subsystem, ~~train~~ division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). A.1

~~OPERATIONAL~~ MODE

~~An OPERATIONAL MODE, i.e.,~~ MODE, shall ~~be~~ correspond to any one inclusive combination of mode switch position, ~~and~~ average reactor coolant temperature, ~~as~~ and reactor vessel head closure bolt tensioning specified in Table ~~1.2~~ 1.1-1 with fuel in the reactor vessel A.10

~~PHYSICS TESTS~~

~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.~~ A.2

<CTS>

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
<1.0> ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
<1.0> 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
<1.0> CHANNEL CALIBRATION all devices in the channel required for channel OPERABILITY and TSTF-205	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 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. C C
<1.0> 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>

1.1 Definitions

<1.0>

CHANNEL CHECK (continued)

status derived from independent instrument channels measuring the same parameter.

<1.0>

CHANNEL FUNCTIONAL TEST

of all devices in the channel required for channel OPERABILITY

TSTF-205

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~~. 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~~.

<1.0>

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.

<1.0>

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.

<1.0>

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)

<CTS>

1.1 Definitions (continued) TSTF-205

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.
MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)	The MFLPD shall be the largest (FLPD) value of the fraction of limiting power density in the core. The FLPD ~~fraction of limiting power density~~ shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.
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).
~~PHYSICS TESTS~~	~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation.~~

required for OPERABILITY

<1.0> <1.0> <1.0> <1.0> <1.0>

3 1 1 2

(continued)

A.1

A.2

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

LCO 3.1.3 All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

A.3 Add proposed ACTIONS Note

A.4 Add proposed Required Action A.1 Note

ACTION:

ACTION A 1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable: A.5

M.1 Add proposed Required Action A.1

L.1 a. Within one hour:

ACTION D 1) Verify that the inoperable control rod, ~~if withdrawn,~~ is separated from all other inoperable ~~withdrawn~~ control rods by at least two control cells in all directions. M.2

L.2 Required Action A.2 2) Disarm the associated ~~directional control valves[a] either:~~ control rod drive (CRD)

~~a) Electrically, or~~ M.3

~~b) Hydraulically by closing the drive water and exhaust water isolation valves.~~ LA.1

Add proposed ACTION B M.3

ACTION E b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

SR 3.1.3.2 SR 3.1.3.3 1. When above the low power setpoint of the RWM, all withdrawn control rods ~~not required to have their directional control valves disarmed electrically or hydraulically~~ shall be demonstrated OPERABLE by moving each control rod at least one notch: A.6 M.5

SR 3.1.3.2 Note

SR 3.1.3.2 a. At least once per 7 days[b] for each fully withdrawn control rod, and at

SR 3.1.3.3 least once per 31 days[b] for each partially withdrawn control rod, and

SR 3.1.3.3 Note

Required Action A.3 b. Within 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.

A.7

~~2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.F, 4.3.G, 4.3.H and 4.3.I.~~

A.8

~~a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.~~

SR 3.1.3.2 and SR 3.1.3.3 b Not required to be performed until 7 days (for fully withdrawn) or 31 days (for partially withdrawn) after the control rod is withdrawn and above the low power setpoint of the RWM.

A.1

ITS 3.1.3

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

Required Action A.4

c. Comply with Surveillance Requirement 4.3.A.2 within ~~24~~ 72 hours ~~or~~ be in HOT SHUTDOWN within the next 12 hours. L.4

ACTION E

ACTION C

2. With one or more control rods scrammable but inoperable for causes other than addressed in ACTION 3.3.C.1 above:

add proposed Note to Condition D L.1

a. If the inoperable control rod(s) is withdrawn, within one hour:

L.1

ACTION D

1) Verify that the inoperable ~~withdrawn~~ control rod(s) is separated from all other inoperable ~~withdrawn~~ control rods by at least two control cells in all directions, and M.2

A.4

add proposed Required Action C.1 Note

L.5

Required Actions C.1 and C.2

2) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the inoperable withdrawn control rod(s) at least one notch by drive water pressure within the normal operating range.[b] M.6

b. With the provisions of ACTION 2.a above not met, fully insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves[a] either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

CRA LA.1

M.6

b The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

A.8

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION | 4.3 - SURVEILLANCE REQUIREMENTS

L.5

Required Action C.2

c. If the inoperable control rod(s) is fully inserted, within one 4 hour disarm the associated directional CRD control valves[a] either:

LA.1

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

ACTION E 3. With the provisions of ACTION 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

ACTION E 4. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

A.8

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

ITS 3.1.3

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

2) Hydraulically by closing the drive water and exhaust water isolation valves.

LA.1

ACTION E

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours:

a. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing control rod and demonstrating that the control rod will not go to the overtravel position, or

b. If recoupling is not accomplished, declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] within one hour, either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

L.7

L.7

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

L.7

3.3 - LIMITING CONDITIONS FOR OPERATION | 4.3 - SURVEILLANCE REQUIREMENTS

ACTION E 2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod position indicator inoperable:

   a. Move the control rod to a position with an OPERABLE position indicator, or

   b. Fully insert the control rod.

A.16 moved to ITS 3.9.4

A.16 moved to ITS 3.9.4

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

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). Additionally, for General Electric fuel types in the Unit 2 core, APLHGR limits are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs).
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, 4, and 5. LOCA analyses are performed to ensure that the determined 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 References 1 and 5. 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. For Unit 2 GE fuel, the APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by the minimum anticipated local peaking factor. For Unit 1 GE fuel and all Siemens Power Corporation fuel, APLHGR limits are typically set high enough such that the LHGR limits are more limiting than the APLHGR limits. For single recirculation loop operation, a conservative multiplier is applied to the exposure dependent APLHGR limits for two loop operation (Ref. 6). This additional limitation is due to the conservative analysis assumption of

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA. For GE fuel types in the Unit 2 core, the APLHGR limits also incorporate the results of the fuel design limits. The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1, 2, 3 and 4. 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 to ensure adherence to fuel design limits during the limiting AOOs (Ref. 4). The APLHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. For two recirculation loops operating, the limit is dependent on exposure. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by a conservative multiplier determined by a specific single recirculation loop analysis (Ref. 6).
APPLICABILITY	The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Studies and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

(continued)

All changes are [1] unless otherwise indicated

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 limits~~ specified in 10 CFR 50.46.

Additionally, for General Electric fuel types in the Unit 2 core, APLHGR limits are specified to ensure

Criteria

are met

2, 3

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 ~~and 2~~.

4

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~~.

5

move to next page as indicated

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~~).

move to next page as indicated

4

~~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)

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow MAPFAC$_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by MAPFAC$_p$ and MAPFAC$_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference 9.~~

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. For Unit 2 GE fuel,

LOCA analyses are ~~then~~ performed to ensure that ~~the above determined~~ APLHGR limits are adequate to meet the ~~PCT~~ peak cladding temperature 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 ~~10~~ 1 and 5. 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. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by ~~its~~ the minimum anticipated local peaking factor. ~~A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.~~

For Unit 1 GE fuel and all Siemens Power Corporation fuel, APLHGR limits are typically set high enough such that the LHGR limits are more limiting than the APLHGR limits.

$\boxed{2}$

For single recirculation loop operation, ~~the MAPFAC multiplier is limited to a maximum of 0.75 (Ref. 5)~~ a conservative multiplier is applied to the exposure dependent APLHGR limits for two loop operation (Ref. 6). This ~~maximum limit~~ 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.

For GE fuel types in the Unit 2 core, the APLHGR limits also incorporate the results of the fuel design limits.

move from ASA first paragraph

move from ASA second paragraph

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

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. For two recirculation loops operating, the limit is ~~determined by multiplying the smaller of the MAPFAC$_p$ and MAPFAC$_f$ factors times the exposure dependent APLHGR limits~~ dependent on exposure. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating,"

(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 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. These SRs apply to each Function in Table 3.3.3.1-1, except where identified in the SR.

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 other required channel in the associated Function is OPERABLE.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1.1	Perform CHANNEL CHECK.	31 days
SR 3.3.3.1.2	Perform CHANNEL CALIBRATION for Functions 7 and 8.	92 days
SR 3.3.3.1.3	Perform CHANNEL CALIBRATION for Functions other than Functions 7 and 8.	24 months

Table 3.3.5.1-1 (page 3 of 4)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	
3. High Pressure Coolant Injection (HPCI) System						
a. Reactor Vessel Water Level − Low Low	1, $2^{(c)}$, $3^{(c)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.9	≥ -56.78 inches	A
b. Drywell Pressure − High	1, $2^{(c)}$, $3^{(c)}$	4	B	SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.9	≤ 2.43 psig	A
c. Reactor Vessel Water Level − High	1, $2^{(c)}$, $3^{(c)}$	2	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.9	≤ 54.23 inches	A
d. Contaminated Condensate Storage Tank (CCST) Level − Low	1, $2^{(c)}$, $3^{(c)}$	2	D	SR 3.3.5.1.4 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 598 ft 1 inch	B
e. Suppression Pool Water Level − High	1, $2^{(c)}$, $3^{(c)}$	2	D	SR 3.3.5.1.4 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 15 ft 11.25 inches	A C
f. High Pressure Coolant Injection Pump Discharge Flow − Low (Bypass)	1, $2^{(c)}$, $3^{(c)}$	1	E	SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.9	≥ 634 gpm	A
g. Manual Initiation	1, $2^{(c)}$, $3^{(c)}$	1	C	SR 3.3.5.1.9	NA	
4. Automatic Depressurization System (ADS) Trip System A						
a. Reactor Vessel Water Level − Low Low	1, $2^{(c)}$, $3^{(c)}$	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.9	≥ -56.78 inches	A
b. Drywell Pressure − High	1, $2^{(c)}$, $3^{(c)}$	2	F	SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.9	≤ 2.43 psig	A
c. Automatic Depressurization System Initiation Timer	1, $2^{(c)}$, $3^{(c)}$	1	G	SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 119 seconds	B

(continued)

(c) With reactor steam dome pressure > 150 psig.

Table 3.3.5.2-1 (page 1 of 1)
Reactor Core Isolation Cooling System Instrumentation

FUNCTION	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	
1. Reactor Vessel Water Level – Low Low	4	B	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.6	$\geq$ -56.78 inches	A
2. Reactor Vessel Water Level – High	2	C	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.6	$\leq$ 54.23 inches	A
3. Contaminated Condensate Storage Tank (CCST) Level – Low	2	D	SR 3.3.5.2.4 SR 3.3.5.2.5 SR 3.3.5.2.6	$\geq$ 598 ft 1 inch	B
4. Suppression Pool Water Level – High	2	D	SR 3.3.5.2.4 SR 3.3.5.2.5 SR 3.3.5.2.6	$\leq$ 15 ft 11.25 inches	B C
5. Manual Initiation	1	C	SR 3.3.5.2.6	NA	

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	≥ -55.2 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	≤ 0.331 seconds
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 3.3.6.1.6 SR 3.3.6.1.7	≤ 138% rated steam flow
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	≤ 198°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	≥ 11.8 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	≤ 2.43 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	≤ 70 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	≤ 151% rated steam flow
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	≥ 113.0 psig
d. Drywell Pressure − High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 2.43 psig
e. HPCI Turbine Area Temperature − High	1,2,3	2	F	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 169°F
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line 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	≤ 175% rated steam flow
b. RCIC 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. RCIC Steam Supply Line Pressure − Low	1,2,3	4[a]	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≥ 54 psig
d. RCIC Turbine Area Temperature − High	1,2,3	2	F	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 169°F

(continued)

(a) Only inputs into one trip system.

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	≥ 11.8 inches
6. RHR Shutdown Cooling System Isolation					
a. Reactor Vessel Pressure − High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 130 psig
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	≥ 11.8 inches

(b) In MODES 4 and 5, provided RHR 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

8. Turbine Stop Valve — Closure (continued)

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 ≥ 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.

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 ≥ 45% RTP.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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 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 ≥ 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.

10. Turbine Condenser Vacuum—Low

The Turbine Condenser Vacuum—Low Function is provided to shut down the reactor and reduce the energy input to the main condenser to help prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. The Turbine Condenser Vacuum—Low Function is the primary scram signal for the loss of condenser vacuum event analyzed in Reference 9. For this event, the reactor scram reduces the amount of energy required to be absorbed by the main condenser and helps to ensure the MCPR SL is not exceeded by reducing the core energy prior to the fast closure of the turbine stop valves. This Function helps maintain the main condenser as a heat sink during this event.

Turbine condenser vacuum pressure signals are derived from four pressure switches that sense the pressure in the condenser. The Allowable Value was selected to reduce the

(continued)

BASES

ACTIONS (continued)

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. C

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 13) to permit restoration of any inoperable required channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a scram), Condition D must be entered and its Required Action taken. The 12 hour allowance is not allowed for Reactor Mode Switch — Shutdown Position and Manual Scram Function channels since with one channel inoperable RPS trip capability is not maintained. In this case, Condition C must be entered and its Required Actions taken.

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic, for any Function, would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in Reference 13 for the 12 hour Completion Time. Within the 6 hour allowance, the associated Function will have all required channels OPERABLE or in trip (or any combination) in one trip system.

Completing one of these Required Actions restores RPS to a reliability level equivalent to that evaluated in Reference 13, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels if the two inoperable channels are in the same Function while the four

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.2 (continued)

consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR, APLHGR, and LHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

The Average Power Range Monitor Flow Biased Neutron Flux—High Function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the total loop drive flow signals from the flow converters used to vary the setpoint is appropriately compared to a calibrated flow signal and, therefore, the APRM Function accurately reflects the required setpoint as a function of flow. Each flow signal from the respective flow converter must be ≤ 100% of the calibrated flow signal. If the flow converter signal is not within the limit, all required APRMs that receive an input from the inoperable flow converter must be declared inoperable.

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.4 and SR 3.3.1.1.8 (continued)

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 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.5 (continued)

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.

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.

(continued)

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 3.3.1.1.9

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 and SR 3.3.1.1.15

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 92 day Frequency of SR 3.3.1.1.10 is based on the reliability analysis of Reference 13. The 24 month Frequency of SR 3.3.1.1.15 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 3.3.1.1.11

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 3.3.1.1.12, 3.3.1.1.14, and SR 3.3.1.1.16

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 3.3.1.1.14 and SR 3.3.1.1.16 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 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.12, 3.3.1.1.14, and SR 3.3.1.1.16 (continued)

complete the SR. Note 3 to SR 3.3.1.1.14 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 3.3.1.1.16.

The Frequency of SR 3.3.1.1.12 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 3.3.1.1.14 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 3.3.1.1.16 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 3.3.1.1.13

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 $\geq$ 45% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into 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 $\geq$ 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 $\geq$ 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.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.13 (continued)

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

SR 3.3.1.1.17

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.

SR 3.3.1.1.18

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.18 (continued)

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.
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. B
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). C

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.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.7 (continued)

Note 2 to SR 3.3.1.2.6 allows the Surveillance to be delayed until entry into the specified condition of the 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 ≤ 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 ≤ 10% RTP in MODE 1. The Note to SR 3.3.2.1.2 allows entry into MODE 2 on a startup and entry in MODE 2 concurrent with a power reduction to ≤ 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 ≤ 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.

C

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 3.3.1.1.8.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.5

The RBM is automatically bypassed when power is below a specified value or if a peripheral control rod is selected. The power level is determined from the APRM signals input to each RBM channel. The automatic bypass setpoint must be verified periodically to be < 30% RTP. In addition, it must also be verified that the RBM is not bypassed when a control rod that is not a peripheral control rod is selected (only one non-peripheral control rod is required to be verified). If any bypass setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the APRM channel can be placed in the conservative condition to enable the RBM. If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8. 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.7 (continued)

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 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.

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.

(continued)

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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.2.1 (continued)

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

SR 3.3.2.2.3

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 3.3.2.2.3 (continued)

The Frequency is based upon the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the 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.

LCO

4. Drywell Pressure (continued)

pressure signals are transmitted 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 31 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)

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 8 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

(continued)

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

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 (> 1 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.

High Pressure Coolant Injection System

The HPCI System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level — Low Low or Drywell Pressure — High. The Reactor Vessel Water Level — Low Low variable is monitored by four redundant differential pressure switches and the Drywell Pressure — High variable is monitored by four redundant pressure switches. The output of each switch is connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function. The logic can also be initiated by use of a Manual Initiation push button.

The HPCI pump discharge flow is monitored by a differential pressure switch. When the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.

(continued)

BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

The HPCI full flow test line isolation valves are closed upon receipt of a HPCI initiation signal to allow the full system flow assumed in the accident analysis.

The HPCI System also monitors the water levels in the two contaminated condensate storage tanks (CCSTs) and the unit suppression pool because these are the two sources of water for HPCI operation. Reactor grade water in the CCSTs is the normal source. The HPCI System is normally aligned to both CCSTs. Upon receipt of a HPCI initiation signal, the CCST suction valve is automatically signaled to open (it is normally in the open position) unless both pump suction valves from the suppression pool are open. If the water level in any CCST falls below a preselected level, first the suppression pool suction valves automatically open, and then when the valves are fully open the CCST suction valve automatically closes. Two level switches are used to detect low water level in each CCST. The outputs for these switches are provided to logics of HPCI in both Unit 1 and Unit 2. Any switch can cause the suppression pool suction valves to open and the CCST suction valve to close. The suppression pool suction valves also automatically open and the CCST suction valve closes if high water level is detected in the suppression pool (one-out-of-two logic). To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level — High trip, at which time the HPCI turbine trips, which causes the turbine's stop valve and the pump discharge valve to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level — Low Low signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated by either automatic or manual means, although manual initiation requires manipulation of each individual relief valve control switch. Automatic

(continued)

BASES

BACKGROUND

<u>Automatic Depressurization System</u> (continued)

initiation occurs when signals indicating Reactor Vessel Water Level – Low Low, Drywell Pressure – High, CS or LPCI Pump Discharge Pressure – High are all present and the ADS Initiation Timer has timed out. ADS automatic initiation also occurs when signals indicating Reactor Vessel Water Level – Low Low are present and the ADS Low Low Water Level Actuation Timer times out. However, this initiation occurs since this logic provides a direct initiation of the associated low pressure ECCS pumps, thereby bypassing the CS or LPCI Reactor Steam Dome Pressure (Permissive) channels. After the pumps start the ADS Drywell Pressure – High contacts are effectively bypassed and the above logic is completed after CS or LPCI Pump Discharge Pressure – High channels are actuated and the ADS Initiation Timer has also timed out. There are two differential pressure switches for Reactor Vessel Water Level – Low Low and two pressure switches for Drywell Pressure – High, in each of the two ADS trip systems. Each of these switches connects to a relay whose contacts form the initiation logic.

Each ADS trip system includes time delays between satisfying the initiation logic and the actuation of the ADS valves. The ADS Initiation Timer time delay setpoint and the Low Low Water Level Actuation Time Delay Setpoint are chosen to be long enough that the HPCI has sufficient operating time to recover to a level above Low Low, yet not so long that the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the timers is timing. Resetting the ADS initiation signals resets the ADS Initiation Timers.

The ADS also monitors the discharge pressures of the four LPCI pumps and the two CS pumps. Each ADS trip system includes two discharge pressure permissive switches from all CS and LPCI pumps. However, only the switches in the associated division are required to be OPERABLE for each trip system (i.e., Division 1 LPCI pumps A and B input to ADS trip system A, and Division 2 LPCI pumps C and D input to ADS trip system B). The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. Any one of the six low pressure pumps is sufficient to permit automatic depressurization.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Allowable Values are specified for each ECCS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. 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.

Some Functions (i.e, Functions 1.c, 1.d, 2.c, 2.d, 4.d, 4.e, 5.d, and 5.e) have both an upper and lower analytic limit that must be evaluated. The Allowable Values and trip setpoints are derived from both an upper and lower analytic limit using the methodology describe above. Due to the upper and lower analytic limits, Allowable Values of these Functions appear to incorporate a range. However, the upper and lower Allowable Values are unique, with each Allowable Value associated with one unique analytic limit and trip setpoint.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.1 (continued)

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 3.3.5.1.2 and SR 3.3.5.1.4

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 31 days for SR 3.3.5.1.2 is based on engineering judgement and the reliability of the equipment. The Frequency of 92 days for SR 3.3.5.1.4 is based on the reliability analyses of Reference 4.

SR 3.3.5.1.3, SR 3.3.5.1.6, SR 3.3.5.1.7, and SR 3.3.5.1.8

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 3.3.5.1.3 is based upon the assumption of a 60 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.3, SR 3.3.5.1.6, SR 3.3.5.1.7, and SR 3.3.5.1.8 (continued)

Frequency of SR 3.3.5.1.6 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 3.3.5.1.7 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 3.3.5.1.8 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 3.3.5.1.5

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 4.

SR 3.3.5.1.9

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.3.5.1.9 (continued) 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. NEDC-30936-P-A, "BWR Owners' Group Technical Specification Improvement Analyses for ECCS Actuation Instrumentation, Part 1 and Part 2," December 1988.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.2.1 (continued)

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 and SR 3.3.5.2.4

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 31 day Frequency of SR 3.3.5.2.2 is based on the reliability of the components. The 92 day Frequency of SR 3.3.5.2.4 is based on the reliability analysis of Reference 1.

SR 3.3.5.2.3 and SR 3.3.5.2.5

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.2.3 and SR 3.3.5.2.5 (continued)

The Frequency of SR 3.3.5.2.3 is based upon the assumption of a 60 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.5.2.5 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 3.3.5.2.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.3 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-770-06-2A, "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, the Main Steam Line Pressure — Low, and the 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.

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 Reactor Core Isolation Cooling System Isolation (continued)

HPCI and RCIC 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 strings. 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. 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 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. | C

RWCU Functions isolate the Group 3 valves.

6. Residual Heat Removal (RHR) 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 strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the RHR 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 Reactor Vessel Pressure — High Function receives input from two channels, both of which provide input to both trip systems. Any | C

(continued)

BASES

BACKGROUND	<u>6. Residual Heat Removal (RHR) Shutdown Cooling (SDC) System Isolation</u> (continued) 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 RHR SDC suction penetration. Shutdown Cooling System Isolation Functions isolate some Group 2 valves (RHR SDC suction 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 3 and 4 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. The OPERABILITY of the primary containment instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.6.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time, where appropriate. Allowable Values are specified for each Primary Containment Isolation Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Main Steam Line Pressure - Low (continued)

failure (Ref. 6). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

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

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

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 6).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Pressure - Timer

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 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. 6). 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.1 (continued)

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.

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analyses described in References 9 and 10. 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.3 (continued)

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.

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. UFSAR, Table 6.2-7.
2. 10 CFR 50.62.
3. UFSAR, Section 6.2.

(continued)

BASES

REFERENCES (continued)

4. UFSAR, Chapter 15.

5. UFSAR, Section 15.6.5.

6. UFSAR, Section 15.1.3.

7. UFSAR, Section 15.6.4.

8. UFSAR, Section 9.3.5.

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

10. 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. C

(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.

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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 15.7.2.
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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.1

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

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

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

SR 3.3.7.1.2

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1.2 (continued)

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

The Frequency of 92 days is based on the reliability analyses of Reference 4.

SR 3.3.7.1.4 and SR 3.3.7.1.5

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

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

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.6

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

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

REFERENCES

1. UFSAR, Section 6.4.
2. UFSAR, Section 15.6.4.
3. UFSAR, Section 15.6.5.
4. 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

SURVEILLANCE REQUIREMENTS

SR 3.3.7.2.1 (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 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 the required channels of this LCO.

SR 3.3.7.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. 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 4.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.2.3 and SR 3.3.7.2.4

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. A Note to SR 3.3.7.2.3 states that radiation detectors are excluded from CHANNEL CALIBRATION since they are calibrated in accordance with SR 3.3.7.2.4.

The Frequency of SR 3.3.7.2.3 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift associated with the channel, except for the radiation detectors, in the setpoint analysis. The Frequency of SR 3.3.7.2.4 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift for the radiation detector in the setpoint analysis.

SR 3.3.7.2.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the mechanical vacuum pump breakers and isolation valve 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 or the isolation valve 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.

(continued)

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)

SR 3.3.8.1.5

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 8.3.1.8.
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

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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 7.2.2.
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

TABLE ~~4.1.A-1~~ 3.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

Note 1 SR 3.3.1.1.14 SR 3.3.1.1.16 (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~~ LA.3

SR 3.3.1.1.7 determined to overlap ~~for at least (½) decades~~ during each controlled shutdown, if not performed within the previous 7 days. LA.3

SR 3.3.1.1.4 (c) ~~Within 24 hours prior to startup, if not performed within the previous 7 days.~~ The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement. A.12

SR 3.3.1.1.2 (d) 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. 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 L.7

ACTIONS Note 2

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.~~ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP

SR 3.3.1.1.2 Note

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

SR 3.3.1.1.11 SR 3.3.1.1.16 (h) Trip units are calibrated at least once per ~~31~~ 92 days and transmitters are calibrated at the frequency identified in the table. A.3

A.6

(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 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

REFUELING OPERATIONS

Instrumentation 3/4.10.B

3.10 - LIMITING CONDITIONS FOR OPERATION

LCO 3.3.1.2 and Table 3.3.1.2-1

B. Instrumentation

At least 2 source range monitor[a] (SRM) CHANNEL(s) shall be OPERABLE ~~and inserted to the normal operating level with:~~

~~1. Continuous visual indication in the control room, and~~ LA.2

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.3 Add proposed SR 3.3.1.2.2.a and Note 2

APPLICABILITY:

OPERATIONAL MODE 5~~, unless the following conditions are met:~~

SR 3.3.1.2.4 Note

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

M.2

M.5 and determination of signal to noise ratio

Add proposed SR 3.3.1.2.4 Note

L.4

L.8 or ≥ 0.7cps with a signal to noise ratio ≥ 20:1

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.

LA.2

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

SR 3.3.1.2.2 b and c Note 1

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

M.5 add proposed SR 3.3.1.2.5 Note

SR 3.3.1.2.5 2. Performance of a CHANNEL FUNCTIONAL TEST:

L.3

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

b. At least once per 7 days.

SR 3.3.1.2.4 3. Verifying that the channel count rate is at least 3 cps:

~~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.

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

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.

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 $\leq$ 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 $\geq$ 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 $\leq$ 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. 6

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 $\geq$ 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 A

TECHNICAL CHANGES - LESS RESTRICTIVE

LF.1 (cont'd) 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 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. A A

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. A

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

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) Specifications Improvement Analysis for BWR Control Rod Block 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 < 10% 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 reactor is operating on a LIMITING CONTROL ROD PATTERN, have been

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd) 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.

RELOCATED SPECIFICATIONS

R.1 The SRM, IRM, Scram Discharge Volume, and APRM control rod blocks of CTS 3/4.2.E function to prevent a 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 Quad Cities 1 and 2 Technical Specifications and have been relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the Quad Cities 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled in accordance with 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

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.

L.6 CTS Table 3.2.F-1 and Table 4.2.F-1 Applicability requirement for Drywell Area Radiation Monitors, during MODES 1, 2, and 3 is proposed to be changed to MODES 1 and 2. Proposed ITS 3.3.3.1 Applicability requires PAM instrumentation only in MODES 1 and 2. These instruments should not be required in MODE 3 because they are required to monitor variables related to the diagnosis and preplanned actions required to mitigate design basis accidents occurring in MODES 1 and 2. In MODES 3, 4, and 5, plant conditions are such that the likelihood of an event that would require PAM instrumentation is extremely low. Therefore, the PAM instrumentation is not required to be OPERABLE in MODES 3, 4, and 5. This change is consistent with BWR ISTS, NUREG-1433, Revision 1.

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. B

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 documented 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 1151DP4 Transmitters, Amerace ETR14B3CC2004003 time delay relays, and General Electric Model 184C5988G131 Analog Trip Units. The General Electric Analog 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 General Electric Analog 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 sufficient quantity of As Found and As Left calibration

TECHNICAL CHANGES - LESS RESTRICTIVE

LE.1 (cont'd) data was not available to perform a rigorous drift analysis for the time delay relays. The vendor's drift allowance was determined per NES-EIC-20.04, Rev. 2, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy" and used to calculate a 30 month drift. The calculated 30 month drift was used in the development of the plant setpoint and the Technical Specification Allowable Value. The results of this analysis support a 24 month surveillance interval.

Functional Unit 2, Reactor Vessel Pressure - High

This function is performed by Rosemount 1151GP9 Transmitters and General Electric Model 184C5988G131 Trip Units (existing Rosemount trip units scheduled for replacement with the General Electric Trip Units during Q1R16 for Unit 1). The General Electric 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 General Electric Analog 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LF.1 (cont'd) (Nuclear Engineering Standard NES-EIC-20.04, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy"). For most cases, the Allowable Value determinations were calculated using plant specific operating and surveillance trend data or an allowance as provided for by the Instrument Setpoint Methodology. For all other cases, vendor documented performance specifications for drift were used. The Allowable Value verification used actual plant operating and surveillance trend information to ensure the validity of the developed Allowable Value. All changes to safety analysis limits applied in the methodologies were evaluated and confirmed as ensuring safety analysis licensing acceptance limits are maintained. All design limits applied in the methodologies were confirmed as ensuring that applicable design requirements of the associated systems and equipment are maintained. The methodologies used have been compared with the guidance of ANSI/ISA S67.04-Part I-1994 and ANSI/ISA RP67.04-Part II-1994. Plant calibration procedures will ensure that the assumptions regarding calibration accuracy, measurement and test equipment accuracy, and setting tolerance are maintained.

Setpoints for each design or safety analysis limit have been established by accounting for the applicable instrument accuracy, calibration and drift uncertainties, environmental effects, power supply fluctuations, as well as uncertainties related to process and primary element measurement accuracy using the Instrument Setpoint Methodology. The Allowable Values have been established from each design or safety analysis limit 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"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.

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.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd)

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.

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. C

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd)

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 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

TABLE 3.2.B-1 (Continued)

A.8

ECCS ACTUATION INSTRUMENTATION

Insert ACTION 31

ACTION

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.~~

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

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

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

L.1

or declare associated ECCS pump inop.

ACTION 32 - 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.

ACTION B

ACTION H

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 - 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~~ 24 hours or declare the associated ECCS system(s) inoperable.

ACTION C

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.

Insert ACTION 37 and 38

A.8

3.3.5.1-1
TABLE ~~4.2.B-1~~

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

M.1 add Core Spray Pump Time Start - Time Delay Relay (Function 1.e)

	Functional Unit	SR 3.3.5.1.1 CHANNEL CHECK	SR 3.3.5.1.2 SR 3.3.5.1.4 CHANNEL FUNCTIONAL TEST	SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6 SR 3.3.5.1.8 CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1	1. CORE SPRAY (CS) SYSTEM				Table 3.3.5.1-Note (a)
1.a	a. Reactor Vessel Water Level - Low Low	S - 1	M - 2	A.8 E - B/M - 3	1, 2, 3, 4(a), 5(a)
1.b	b. Drywell Pressure - High(b) A.6	NA	A.8 M - Q - 4	Q - 6	1, 2, 3
1.c	c. Reactor Vessel Pressure - Low (Permissive)	NA	M - Q - 4	Q - 6	1, 2, 3, 4(a), 5(a)
1.d	d. CS Pump Discharge Flow - Low (Bypass)	NA	M - Q - 4	8 - E - 5 24 months LE.1	1, 2, 3, 4(a), 5(a) A
	2. LOW PRESSURE COOLANT INJECTION (LPCI) SUBSYSTEM			A.8	Table 3.3.5.1 Note (a)
2.a	a. Reactor Vessel Water Level - Low Low	S - 1	M - 2	E - B/M - 3	1, 2, 3, 4(a), 5(a)
2.b	b. Drywell Pressure - High(b) A.6	NA	A.8 M - Q - 4	Q - 6	1, 2, 3 A.1
2.c	c. Reactor Vessel Pressure - Low (Permissive)	NA	M - Q - 4	Q - 6	1, 2, 3, 4(a), 5(a)
2.f	d. LPCI Pump Discharge Flow - Low (Bypass)	NA	M - Q - 4	8 - E - 5 24 months LE.1	1, 2, 3, 4(a), 5(a) A
	Table 3.3.5.1 Note (c) 3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM(d)			8 A.8 24 months	
3.a	a. Reactor Vessel Water Level - Low Low	S - 1	M - 2	E - B/M - 3	1, 2, 3 A
3.b	b. Drywell Pressure - High(b) A.6	NA	M - Q - 4	Q - 6	1, 2, 3
3.d	c. Condensate Storage Tank Level - Low	NA	A.8 M - Q - 4	M.5 NA	1, 2, 3
3.e	d. Suppression Chamber Water Level - High	NA	M - Q - 4	NA	1, 2, 3
3.c	e. Reactor Vessel Water Level - High (Trip)	NA	M - 2	A.8 E - B/M - 3	1, 2, 3
3.f	f. HPCI Pump Discharge Flow - Low (Bypass)	NA	A.8 M - Q - 4	6 - E	1, 2, 3
3.g	g. Manual Initiation	NA	SR 3.3.5.1.8 - E - A.11 LD.1	NA Q M.3	1, 2, 3 A

add Functions 2.d, 2.e, 2.g, 2.h, 2.i, 2.j and 2.k M.1

ADMINISTRATIVE (continued)

A.9 The technical content of the requirements of CTS Table 3.2.B-1 Functional Units 6.a and 6.b and Table 4.2.B-1 Functional Units 5.a and 5.b, including associated Notes and Actions, are being moved to ITS 3.3.8.1, "Loss of Power Instrumentation," in accordance with the format of the BWR ISTS, NUREG-1433, Rev. 1. Any technical changes to these requirements are addressed in the Discussion of Changes for ITS: 3.3.8.1, in this Section.

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 3.3.5.1.8 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

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 3.3.5.1.4, SR 3.3.5.1.7, SR 3.3.5.1.8 and SR 3.3.5.1.9, 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 B

M.2 A maximum Allowable Value has been added for the CS 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.

M.3 CTS Table 4.2.B-1 Functional Unit 3.f requires the performance of a CHANNEL CALIBRATION of the HPCI Pump Discharge Flow - Low (Bypass) once per 18 months. ITS Table 3.3.5.1-1 Function 3.f requires the performance of a CHANNEL CALIBRATION once per 92 days (SR 3.3.5.1.6). This change is consistent with current plant practice. The change represents an additional restriction on plant operation since the more restrictive surveillance frequency of 92 days will be included in Technical Specifications. This change is necessary to ensure the associated instrumentation is maintained OPERABLE. A

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.4 Not used.

M.5 CTS Table 4.2.B-1 requires a CHANNEL FUNCTIONAL TEST (CFT) of Functional Unit 3.c, Condensate Storage Tank Level - Low, and 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 these Functions 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 3.3.5.1.8) at a Frequency of 24 months consistent with drift analysis assumptions in the plant setpoint methodology.

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 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LF.1 (cont'd) ensure the validity of the developed Allowable Value. All changes to safety analysis limits applied in the methodologies were evaluated and confirmed as ensuring safety analysis licensing acceptance limits are maintained. All design limits applied in the methodologies were confirmed as ensuring that applicable design requirements of the associated systems and equipment are maintained. The methodologies used have been compared with the guidance of ANSI/ISA S67.04-Part I-1994 and ANSI/ISA RP67.04-Part II-1994. Plant calibration procedures will ensure that the assumptions regarding calibration accuracy, measurement and test equipment accuracy, and setting tolerance are maintained. Setpoints for each design or safety analysis limit have been established by accounting for the applicable instrument accuracy, calibration and drift uncertainties, environmental effects, power supply fluctuations, as well as uncertainties related to process and primary element measurement accuracy using the Instrument Setpoint Methodology. The Allowable Values have been established from each design or safety analysis limit 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 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, C

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) 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

3.3.5.2-1

TABLE 4.2.D-1

REACTOR CORE ISOLATION COOLING ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	SR 3.3.5.2.1 CHANNEL CHECK	SR 3.3.5.2.2 SR 3.3.5.2.4 CHANNEL FUNCTIONAL TEST	SR 3.3.5.2.3 SR 3.3.5.2.5 CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low	S -1	M - 2	A.4 E B/M - 3
2. Reactor Vessel Water Level - High (Trip)	S - 1	M- 2	E B/M - 3
3. Condensate Storage Tank Level - Low	~~NA~~	A.4 M Q - 4	NA 24 months - 5 D.1
4. Suppression Chamber Water Level - High	~~NA~~	M Q - 4	NA
5. Manual Initiation	~~NA~~	A.7 E - 6 24months LD.1	~~NA~~ M.2

DISCUSSION OF CHANGES
ITS: 3.3.5.2 - RCIC SYSTEM INSTRUMENTATION

ADMINISTRATIVE

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

A.2 This proposed change to the CTS 3.2.D Actions provides more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3, "Completion Times," the ITS 3.3.5.2 ACTIONS Note ("Separate Condition entry is allowed for each....") provides direction consistent with the intent of the existing Actions for an inoperable RCIC instrumentation channel. It is intended that each 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.3 The Trip Setpoint for Functional Unit 1, Reactor Water Level - Low Low and Functional Unit 2, Reactor Vessel Level - High (Trip) in Table 3.2.D-1 is referenced to the top of active fuel. The reference value for the Allowable Value specified in ITS Table 3.3.5.2-1 for Functions 1 and 2, 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 changes to the Trip Setpoints are addressed in Discussion of Changes A.8 and LF.1, therefore this change is considered administrative. A

A.4 These changes to CTS 3/4.2.D are provided in the Quad Cities ITS consistent with the Technical Specifications Change Request submitted to the NRC for approval per ComEd letter dated December 27, 1999. The changes identified are consistent with the allowances in GENE-770-06-2-A, "Bases for Changes to Surveillance Test intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specification," December 1992. As such, the changes are administrative.

ADMINISTRATIVE (continued)

A.5 CTS Table 3.2.D-1 Action 42 requires placing the inoperable channel in trip when a Condensate Storage Tank Level—Low or Suppression Chamber Water Level—High channel is inoperable. A new Required Action has been added (ITS 3.3.5.2 Required Action D.2.2) to allow the RCIC pump suction to be aligned to the suppression pool in lieu of tripping the channel, if a Condensate Storage Tank Level—Low or Suppression Chamber Water Level—High channel is inoperable. Since this proposed action results in the same condition as if a channel were tripped (tripping one channel results in the suction being aligned to the suppression pool), this change is considered administrative.

A.6 The column title in CTS Table 3.2.D-1 is on a per Function basis rather than the per Trip System basis indicated in CTS Table 3.2.D-1 Actions 41 and 43. All required channels are specified in the column. Therefore, reference to Trip System has been deleted and replaced with Function as indicated in ITS Table 3.3.5.2-1.

A.7 CTS Table 4.2.D-1 requires a CHANNEL FUNCTIONAL TEST (CFT) of Functional Unit 5, the Manual Initiation Function, every 18 months. CTS 4.2.D.2 and proposed SR 3.3.5.2.6 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 3.3.5.2 only requires an LSFT, and this change is considered administrative.

A.8 CTS 3.2.D requires the RCIC System actuation instrumentation setpoints to be set consistent with the Trip Setpoint values shown in Table 3.2.D-1. CTS 3.2.D 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.D-1. Table 3.2.D-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 3.3.5.2-1). In accordance with current plant procedures and practices, the Trip Setpoints specified in CTS Table 3.2.D-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.D-1 for the RCIC System actuation instrumentation Functions or the Allowable Values specified in ITS Table 3.3.5.2-1 (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.

ADMINISTRATIVE (continued)

A.9 The detail in CTS Table 3.2.D-1 Functional Unit 5, RCIC Manual Initiation, that there is one channel "per system" has been deleted since there is only one RCIC System per unit. Since the Specifications apply equally to Units 1 and 2, this Note is not necessary. Since its removal is editorial, this change is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 Not used.

M.2 CTS Table 4.2.D-1 requires a 92 day CHANNEL FUNCTIONAL TEST of Functional Unit 3, Condensate Storage Tank Level—Low, and Functional Unit 4, Suppression Chamber Water Level — High. The Table does not currently require a CHANNEL CALIBRATION. The channels associated with these Functions include a level switch which must trip at the specified setpoint Allowable Value (see Discussion of Changes A.8 and LF.1). Therefore, a CHANNEL CALIBRATION requirement is added at a Frequency of 24 months consistent with drift analysis assumptions in the plant setpoint methodology. This change represents an additional restriction on plant operation necessary to ensure these RCIC System instruments are maintained OPERABLE.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The detail in CTS Table 3.2.D-1 Note (c) related to the reference setting of the reactor vessel water level instrumentation (CTS Table 3.2.D-1 Functional Units 1 and 2) and the detail for CTS Table 3.2.D-1 for Functional Unit 4 (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 Allowable Value specified in ITS Table 3.3.5.2-1 for the reactor vessel water level instrumentation has been changed to the value associated with "instrument zero," as discussed in Discussion of Change A.3. These details are not necessary to ensure the OPERABILITY of the RCIC System instrumentation. The requirements of ITS 3.3.5.2 and the associated Surveillances are adequate to ensure the RCIC System instrumentation is maintained OPERABLE. Therefore, these relocated details are 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.

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 System design and operation details specified in CTS Table 3.2.D-1, Note (b) (which indicates that the Condensate Storage Tank Level—Low and Suppression Chamber Water Level—High channels provide signals to pump suction valves) are proposed to be relocated to the Bases. Details relating to system design and operation are unnecessary in the LCO. These details are not necessary to ensure the OPERABILITY of the RCIC System Instrumentation. The requirements of ITS 3.3.5.2 and the associated Surveillance Requirements are adequate to ensure the RCIC System 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.D.2 and the CHANNEL FUNCTIONAL TEST for the RCIC Manual Initiation Function specified in CTS Table 4.2.D-1 Functional Unit 5 (changed to LSFT in Discussion Change A.7 above) has been extended from 18 months to 24 months in proposed SR 3.3.5.2.6. This SR ensures that RCIC logic 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. The system function testing performed in ITS 3.5.3 overlaps this surveillance to provide complete testing of the safety function. The RCIC system is tested on a more frequent basis during the operating cycle in accordance with proposed SRs 3.3.5.2.1, 3.3.5.2.2, 3.3.5.2.3, 3.3.5.2.4, and 3.3.5.2.5. This testing of the RCIC system ensures that a significant portion of the RCIC circuitry is operating properly and will detect significant failures of this circuitry. RCIC system actuating logic is designed to be single failure proof and therefore, is highly reliable. Furthermore, as stated in the NRC Safety Evaluation Report (dated

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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. 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 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.

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 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LF.1 (cont'd) 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

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~~ Allowable Values A.7 column.

APPLICABILITY:

As shown in Table 3.2.A-1.

ACTION: add proposed ACTIONS Note A.2

ACTIONS A and B

1. With an isolation actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the ~~Trip Setpoint~~ Allowable Value A.7 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.

A.3 Insert CTS 3.2.A Actions 2

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.~~

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 LD.1 months.

A.3 Insert CTS 3.2.A footnote a

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.~~

TABLE 4.2.A-1 3.3.6.1-1

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)
2. 1. PRIMARY CONTAINMENT ISOLATION		A.3	24 months LE.1	
2.a a. Reactor Vessel Water Level - Low	S-1	M Q-2	6 E(a)-3	1, 2, 3
2.b b. Drywell Pressure - High(b) A.4	NA	M Q-2	Q-4	1, 2, 3
2.c c. Drywell Radiation - High	S-1	M Q-2	E-6	1, 2, 3 A.5
2. SECONDARY CONTAINMENT ISOLATION		A.3		moved to ITS 3.3.6.2 3.3.7.1 A.1
a. Reactor Vessel Water Level - Low(c,d)	S	M Q	E(a)	1, 2, 3 & *
b. Drywell Pressure - High(b,c,d)	NA	M Q	Q	1, 2, 3
c. Reactor Building Ventilation Exhaust Radiation - High(c,d)	S	M Q	Q	1, 2, 3 & **
d. Refueling Floor Radiation - High(c,d)	S	M Q	Q	1, 2, 3 & **
1 3. MAIN STEAM LINE (MSL) ISOLATION		A.3		A.6
1.a a. Reactor Vessel Water Level - Low Low	S-1	M Q-2	6 E(a)-3	1, 2, 3
~~b. MSL Tunnel Radiation - High~~	~~S~~	~~M Q~~	~~E(a)~~	~~1, 2, 3~~
1.b, 1.c c. MSL Pressure - Low	NA	M Q-2	Q-4 24 months-6 for Function 1.c only	1
1.d d. MSL Flow - High(e) A.5	S-1	M Q-2	E-6	1, 2, 3
1.e e. MSL Tunnel Temperature - High	NA	,5 E 24 months LD.1	E-6 24 months LE.1	1, 2, 3

ITS 3.3.6.1

Page 10 of 12

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

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK SR 3.3.6.1.1	CHANNEL FUNCTIONAL TEST SR 3.3.6.1.2 SR 3.3.6.1.5	CHANNEL CALIBRATION SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	Applicable OPERATIONAL MODE(s)
5 4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
5.a a. Standby Liquid Control System Initiation	NA	SR 3.3.6.1.7 E A.10 24 months LD.1	NA	1, 2, 3 L.2
5.b b. Reactor Vessel Water Level - Low	S-1	A.3 M Q-2	6-E (a) 3 24 months LE.1	1, 2, 3
4. 5. REACTOR CORE ISOLATION COOLING ISOLATION				
4.a, 4.b a. Steam Flow - High	NA	A.3 M Q-2	Q-4 24 months-6 for Function 4.b only	1, 2, 3
4.c b. Reactor Vessel Pressure - Low	NA	M Q-2	Q-4	1, 2, 3
4.d c. Area Temperature - High	NA	E 5- 24 months LD.1	E-6	1, 2, 3
3. 6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
3.a, 3.b a. Steam Flow - High	NA	A.3 M Q-2	6 E (a)-3	1, 2, 3
3.c b. Reactor Vessel Pressure - Low	NA	M Q-2	6-E (a)-3	1, 2, 3
3.e c. Area Temperature - High	NA	E	E-6 24 months LE.1	1, 2, 3
add proposed Function 3.d M.2				
6. 7. RHR SHUTDOWN COOLING MODE ISOLATION				
6.b a. Reactor Vessel Water Level - Low	S-1	A.3 M Q-2	6 E (a) 3	3, 4, 5
6.a b. Reactor Vessel Pressure - High (Cut-in Permissive)	NA	M Q-2	Q-4	1, 2, 3

A.1

ITS 3.3.6.1

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

A.5 moved to ITS 3.3.6.2 ITS 3.3.7.1

TABLE NOTATION

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

** When handling irradiated fuel in the secondary containment.

SR 3.3.6.1.3 SR 3.3.6.1.6

92 A.3

(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.

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

(d) Also isolates the control room ventilation system. A.5 moved to ITS 3.3.6.2 ITS 3.3.7.1

(e) ~~These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed every 18 months.~~

A.6

ADMINISTRATIVE (continued)

A.9 The requirement in CTS Table 3.2.A-1 footnote (g) that only one trip system is required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained has been revised for clarification. This footnote is associated with the RHR Shutdown Cooling Mode Isolation Reactor Vessel Water Level—Low Function channels. The logic associated with this sytem is described in the Bases as a one-out-of-two taken twice logic to initiate isolation. Therefore, for trip initiation, one of two channels in each of two trip systems must trip for isolation. The requirements of this footnote have been incorporated in proposed footnote (b) to Table 3.3.6.1-1. The proposed footnote states that in MODES 4 and 5, provided RHR 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. This proposed requirement is consistent with current Technical Specification interpretation of the current requirement and therefore this change is considered administrative. In both cases (in the ITS and CTS), the system will maintain isolation capability at all times.

A.10 The CHANNEL FUNCTIONAL TEST (CFT) requirements for CTS Table 4.2.A-1 Functional Unit 4.a, Standby Liquid Control 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.

A.11 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 terminlogy, 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

ADMINISTRATIVE

A.11 (cont'd) "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.12 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.7 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. B B B

M.2 An additional Function has been added, ITS Table 3.3.6.1-1 Function 3.d. This Function is an additional Drywell Pressure—High Function which isolates the HPCI turbine exhaust vacuum breaker isolation valves coincident with the Reactor Vessel Pressure—Low Function signals. Appropriate ACTIONS and

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

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.10 above), RCIC Area Temperature, and 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:

> "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."

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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 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 the RCIC Steam Flow - 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 1153DB4PA Transmitters and 510DU/710DU 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.

<CTS>

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

<3.1.A>
<2.2.A>

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>
<T 3.1.A-1>
<Appl 2.2.A>

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

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

5

C

C

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

1. Separate Condition entry is allowed for each channel.

<T 4.1.A-1 footnote d>

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.1.A Act 1> <3.1.A Act 2> <3.1.A Act 2.c> <2.2.A Action>	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
<3.1.A Act 2> <3.1.A Act 2.b> <2.2.A Action>	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
<3.1.A Act 2> <3.1.A Act 2.a> <3.1.A Act 3> <2.2.A Action>	C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour

(continued)

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 3.3.1.1.16). In addition, a Note to ISTS SR 3.3.1.1.11 (ITS SR 3.3.1.1.14) 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 Quad Cities 1 and 2 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 6, Drywell Pressure — High
   Function 7, Scram Discharge Volume Water Level — High

   This is consistent with the current licensing basis.

9. The proper Quad Cities 1 and 2 plant specific nomenclature/value/design requirements have been provided.

10. ITS Table 3.3.1.1-1 Function 10, Turbine Condenser Vacuum—Low, has been added consistent with the current licensing basis for RPS Instrumentation. Subsequent Functions 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 Quad Cities 1 and 2 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 Quad Cities 1 and 2 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. C

12. SR 3.3.1.1.8 has been added to the Surveillance Requirements in ITS 3.3.1.1 for the Manual Scram Function. This change is consistent with the CTS.

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 Quad Cities 1 and 2 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 Feedwater System and Main Turbine High Water Level Trip Instrumentation channel logic design is a two-out-of-two for trip actuation. Therefore, with one channel inoperable, the trip logic does not maintain trip capability. As a result, the ISTS 3.3.2.2 allowance for "Separate Condition entry" and ISTS 3.3.2.2 ACTION A have been deleted since these allowances are based on the ability of the instrumentation to perform its safety function upon an actuation of a high water level signal with one channel inoperable. The ISTS 3.3.2.2 Surveillance Requirement Note has been revised consistent with CTS allowances since trip capability is not maintained when any channel is inoperable.

4. ISTS 3.3.2.2 Required Action C.1 (ITS Required Action B.1) 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 (Quad Cities 1 and 2 have three 50% capacity feedwater pumps). Therefore, an additional Required Action is proposed, ITS 3.3.2.2, Required Action B.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 B.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 B.1 has been renumbered due to this addition. C

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

Insert Functions 2g, 2.h, 2.i, 2.j and 2.k 1

<CTS>
<T 3.2.B-1>
<T 4.2.B-1>
<Doc M1>

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
f. Low Pressure Coolant Injection Pump Discharge Flow – Low (Bypass)	1,2,3, 4(a), 5(a)	~~[1 per pump]~~ 1 per loop	E	~~SR 3.3.5.1.1~~ SR 3.3.5.1.5 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.9	≥ 2526 gpm and ≤ [ ] gpm
~~h. Manual Initiation~~	~~1,2,3, 4(a), 5(a)~~	~~[2] [1 per subsystem]~~	~~C~~	~~SR 3.3.5.1.6~~	~~NA~~
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level – Low Low, ~~Level 2~~	1, 2(c), 3(c)	~~4~~	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 ~~SR 3.3.5.1.3~~ SR 3.3.5.1.9 ~~SR 3.3.5.1.7~~	≥ -56.78 inches
b. Drywell Pressure – High	1, 2(c), 3(c)	~~4~~	B	~~SR 3.3.5.1.1~~ SR 3.3.5.1.4 SR 3.3.5.1.6 ~~SR 3.3.5.1.5~~ SR 3.3.5.1.9 ~~SR 3.3.5.1.7~~	≤ 2.43 psig
c. Reactor Vessel Water Level – High ~~Level 8~~	1, 2(c), 3(c)	~~2~~	C	~~SR 3.3.5.1.1~~ SR 3.3.5.1.2 SR 3.3.5.1.3 ~~SR 3.3.5.1.3~~ SR 3.3.5.1.9 ~~SR 3.3.5.1.7~~	≤ 54.23 inches
d. Condensate Storage Tank (CCST) Level – Low	1, 2(c), 3(c)	~~2~~	D	~~SR 3.3.5.1.1~~ SR 3.3.5.1.4 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 598 ft 1 inch
e. Suppression Pool Water Level – High	1, 2(c), 3(c)	~~2~~	D	~~SR 3.3.5.1.1~~ SR 3.3.5.1.4 ~~SR 3.3.5.1.3~~ SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 15 ft 11.25 inches

(continued)

Contaminated

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

(c) With reactor steam dome pressure > ~~[150]~~ psig.

8. (continued)

deleted from the ITS when only the DGs are required to be Operable. This change is also consistent with current licensing basis (CTS Table 3.3.3-1 Footnote * only requires these Functions when the system is required to be Operable per CTS 3.5.B, the ECCS—Shutdown Specification). The DGs are still required to be started on a loss of power signal, as required in ITS 3.3.8.1.

9. 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 or offsite circuit 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 ± 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.

10. The current Quad Cities 1 and 2 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.

11. 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. 1C

12. The word "required" has been added consistent with its use throughout the ITS (not all CCST channels are required in the LCO).

<CTS>

<T 3.2.D-1>
<T 4.2.D-1>

Table 3.3.5.2-1 (page 1 of 1)
Reactor Core Isolation Cooling System Instrumentation

FUNCTION	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level — Low Low, Level 2	~~4~~	B	SR 3.3.5.2.1 SR 3.3.5.2.2 ~~SR 3.3.5.2.3~~ SR 3.3.5.2.3 ~~SR 3.3.5.2.5~~ SR 3.3.5.2.6	≥ ~~-47~~ -56.78 inches
2. Reactor Vessel Water Level — High, Level 8	~~2~~	C	SR 3.3.5.2.1 SR 3.3.5.2.2 ~~SR 3.3.5.2.3~~ SR 3.3.5.2.3 ~~SR 3.3.5.2.5~~ SR 3.3.5.2.6	≤ ~~56.5~~ 54.23 inches
3. Contaminated Condensate Storage Tank (CCST) Level — Low	~~2~~	D	~~SR 3.3.5.2.1~~ SR 3.3.5.2.2 ~~SR 3.3.5.2.3~~ SR 3.3.5.2.4 SR 3.3.5.2.6	≥ 598 ft 1 inch
4. Suppression Pool Water Level — High	~~2~~	D	~~SR 3.3.5.2.1~~ SR 3.3.5.2.2 ~~SR 3.3.5.2.4~~ SR 3.3.5.2.5 SR 3.3.5.2.6	≤ 15 ft 11.25 inches
5. Manual Initiation	~~1~~	C	SR 3.3.5.2.6	NA

<CTS>

SURVEILLANCE REQUIREMENTS

<4.2.A.1>

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

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

<T 3.2.A-1 footnote a>

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
<T 4.2.A-1>	SR 3.3.6.1.1 Perform CHANNEL CHECK.	12 hours
<T 4.2.A-1>	SR 3.3.6.1.2 Perform CHANNEL FUNCTIONAL TEST.	~~[92]~~ days 5
<T 4.2.A-1> <T 4.2.A-1 footnote a>	~~[SR 3.3.6.1.3 Calibrate the trip unit.~~	~~[92]~~ days 5] 5
<T 4.2.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 11] 5
<T 4.2.A-1> <T 4.2.A-1 footnote a>	SR 3.3.6.1.6 Perform CHANNEL CALIBRATION.	~~[18]~~ months 24 5
<4.2.A.2>	SR 3.3.6.1.7 Perform LOGIC SYSTEM FUNCTIONAL TEST.	~~[18]~~ months 24 5

(continued)

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<T 3.2.A-1>
<T 4.2.A-1>
<DOC M.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		5			
a. Reactor Vessel Water Level — Low Low ~~Low, Level 1~~ 7	1,2,3	~~[2]~~	D	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~~	≥ ~~[-129]~~ -55.2 inches 5
b. Main Steam Line Pressure — Low	1	~~[2]~~	E	8 ~~[SR 3.3.6.1.1]~~ ~~[SR 3.3.6.1.2]~~ 5 SR 3.3.6.1.6 SR 3.3.6.1.7 6 ~~SR 3.3.6.1.8~~	≥ ~~[825]~~ 831 psig
~~c.~~ 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 8 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.6 SR 3.3.6.1.7 6 ~~SR 3.3.6.1.8~~	≤ ~~[138]~~ 138 % rated steam flow 5
~~d. Condenser Vacuum — Low~~ 3	~~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 Line Tunnel Temperature — High	1,2,3	~~[2]~~ 2 per trip string	D	9 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.~~2~~ 5 11 8 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.6 SR 3.3.6.1.7 6 ~~SR 3.3.6.1.8~~	≤ ~~[200]~~ 198 °F 5
~~[f. Main Steam Tunnel Differential Temperature — High]~~ 3	~~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 3	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 0.331 seconds

(continued)

~~(a) With any turbine [stop valve] not closed.~~ 3

A C C A C A C B C

<CTS>

<T 3.2.A-1>
<T 4.2.A-1>
<DOC M.4>

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		5			
a. Reactor Vessel Water Level — Low, Level 3 7	1,2,3	~~[2]~~	~~G~~ H 3	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 6 ~~SR 3.3.6.1.8~~	≥ ~~[10]~~ 11.8 5 inches
b. Drywell Pressure — High	1,2,3	~~[2]~~	~~G~~ H	9 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 8 ~~SR 3.3.6.1.3~~ ~~SR 3.3.6.1.6~~ SR 3.3.6.1.7 6 ~~SR 3.3.6.1.8~~ add SR 3.3.6.1.4	≥ ~~[1.92]~~ ≤ 2.43 5 psig 10
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]~~ 70 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	~~[1]~~	F	9 ~~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 6 ~~SR 3.3.6.1.8~~	≤ ~~[303]~~ 151 5 % rated steam flow

(continued)

Insert Function 3.b 3

<CTS>

<DOC M.4>

3 Insert Function 3.b

b. HPCI Steam Line Flow-Timer	1, 2, 3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 3.2 seconds and ≤ 8.8 seconds

B C

<CTS>
<T 3.2.A-1>
<T 4.2.A-1>
<DOC M.2>

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)		5			
c. HPCI Steam Supply Line Pressure — Low	1,2,3	~~[2]~~	F	~~SR 3.3.6.1.1~~ 9 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 6 ~~SR 3.3.6.1.8~~	≥ ~~[100]~~ 113.0 psig 5
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 3
d. Drywell Pressure — High	1,2,3	~~[2]~~ 2	F	9 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 8 ~~[SR 3.3.6.1.3]~~ 10 ~~SR 3.3.6.1.6~~ SR 3.3.6.1.7 7 ~~SR 3.3.6.1.8~~ add SR 3.3.6.14 10	≤ ~~[2.5]~~ 2.43 psig 5
e. HPCI ~~Pipe Penetration Room~~ Turbine Area 7 Temperature — High	1,2,3	~~[2]~~ 2	F	9 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.~~2~~ 5 11 8 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.6 SR 3.3.6.1.7	≤ ~~[169]~~ °F 5
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]

3

(continued)

<CTS>

<T 3.2.A-1>
<T 4.2.A-1>
<DOC M.4>

Table 3.3.6.1-1 (page 4 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. Reactor Core Isolation Cooling (RCIC) System Isolation	Insert Function 4.b 3				
a. RCIC Steam Line Flow — High	1,2,3	5 ~~[1]~~	F	9 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 8 ~~[SR 3.3.6.1.3]~~ 10 ~~SR 3.3.6.1.6~~ SR 3.3.6.1.7 ~~SR 3.3.6.1.8~~ 7 add SR 3.3.6.1.4 10	≤ ~~[300]~~ 175 % rated steam flow 5
3 — c ~~b.~~ RCIC Steam Supply Line Pressure — Low	1,2,3	4(a) ~~[2]~~	F	9 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ 8 10 ~~SR 3.3.6.1.6~~ SR 3.3.6.1.7	~~≥ [60]~~ ≥ 54 psig 5
~~c. RCIC 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~~ 3
~~e. RCIC Suppression Pool Ambient Area 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~~
~~f. 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]~~
~~g. RCIC 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~~
~~h. 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~~

(continued)

(a) Only inputs into one trip system 5

A
C
C
A

<CTS>

<DOC M.4>

3 Insert Function 4.b

b.	RCIC 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

⟨CTS⟩

⟨T 3.2 A-1⟩
⟨T 4.2 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)					
[i. RCIC ~~Equipment Room~~ Turbine Area Temperature — High	1,2,3	~~[1]~~ 2	F	~~[SR 3.3.6.1.1]~~ SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.4 SR 3.3.6.1.7	≤ ~~[ ]~~ 169°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
~~d.~~ a. SLC System Initiation	1,2	~~2~~(b) 1	~~H~~ I	SR 3.3.6.1.7	NA
~~e.~~ b. 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]~~ 11.8 inches
[f. Manual Initiation	1,2,3	[1 per group]	G	SR 3.3.6.1.7	NA]

(continued)

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

L ⟨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. RHR Shutdown Cooling System Isolation					
a. Reactor ~~Steam Dome~~ Vessel Pressure — High	1,2,3	~~[1]~~ 2	F	~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 ~~SR 3.3.6.1.3~~ add SR 3.3.6.1.4 ~~SR 3.3.6.1.6~~ SR 3.3.6.1.7	≤ ~~[145]~~ 130 psig
b. Reactor Vessel Water Level — Low, Level 3	3,4,5	~~[2]~~ (b)	~~J~~ 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]~~ 11.8 inches

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

(b) In MODES 4 and 5, provided RHR 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.

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. Three new Primary Containment Isolation Functions have been added (ITS Table 3.3.6.1-1 Functions 1.c, 3.b and 4.b), consistent with current Quad Cities 1 and 2 Licensing Basis. In addition, 25 Functions have been deleted (ISTS Table 3.3.6.1-1 Functions 1.d, 1.f, 1.g, 1.h, 2.d, 2.e, 2.f, 3.c, 3.f, 3.g, 3.h, 3.i, 3.j, 4.c, 4.d, 4.e, 4.f, 4.g, 4.h, 4.j, 4.k, 5.a, 5.b, 5.c and 5.f) since they are not applicable to Quad Cities 1 and 2. The Functions and ACTIONS have been revised where applicable, to reflect these additions and deletions. 1C

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 Quad Cities 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 Quad Cities 1 and 2 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), Function 3.e (HPCI Turbine Area Temperature—High), and Function 4.d (RCIC Turbine Area Temperature — High), have been revised to reflect current licensing requirements, therefore, SR 3.3.6.1.2 has been replaced with SR 3.3.6.1.5. C

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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve—Closure (continued)

opening

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

B

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~~ even if one TSV fail to should 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%~~ 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.

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 ~~7~~ 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~~ switch 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 OPERABLE~~ the turbine bypass valves ~~must remain shut at THERMAL POWER ≥ 30% RTP~~. opening may affect the OPERABILITY of this Function

8 6

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%~~ 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. 45 3

Insert Function 10

7

11 3

~~10~~. Reactor Mode Switch—Shutdown Position (A3 and B3) 3

two

The Reactor Mode Switch—Shutdown Position Function provides signals, via the 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.

two

The reactor mode switch is a single switch with ~~four~~ 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

converter

accurately reflects the required setpoint as a function of flow. Each flow signal from the respective flow unit must be ≤ 105% of the calibrated flow signal. If the flow unit signal is not within the limit, one required APRM that receives an input from the inoperable flow unit must be declared inoperable.

100 all converter 3

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

TSTF-205

Insert SR 3.3.1.1.4

SR 3.3.1.1.4 and SR 3.3.1.1.8 6

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

2 2

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 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

Twenty four 24 7

7

Insert SR 3.3.1.1.8-1

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).

7 13

Insert SR 3.3.1.1.5

7

~~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~~

TSTF-205 not shown

(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.

7

INSERT SR 3.3.1.1.8-1

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

7
2000 effective full power hours (EFPH)

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.

TSTF-205
Insert SR 3.3.1.1.10

SR 3.3.1.1.9 and SR 3.3.1.1.10 10 5 7

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 3.3.1.1.9 is based on the reliability analysis of Reference 9. 2
13 10 24 of SR 3.3.1.1.15 2 7
3 2

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. 7
24

SR 3.3.1.1.10 11 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

INSERT SR 3.3.1.1.10

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 3.3.1.1.13 (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.

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~~ FSAR 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

(continued)

BASES

1

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2.4

with the detector full in

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.

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.

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

1

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.

SR 3.3.1.2.5 and SR 3.3.1.2.6

Insert SR 3.3.1.2.5

TSTF-205

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.

1

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~~ extended from 7 days to 31 days. The 31 day Frequency is based on operating experience and on other Surveillances (such as

15

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

(continued)

TSTF
-205

INSERT SR 3.3.1.2.5

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)

5

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.]~~

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.

second 4

The Surveillances are modified by a 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. ~~9~~ 10) 1 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

1 "Relay Select Matrix"

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~~ System input.

TSTF-205 Insert SR

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). 1

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

(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 3.3.2.1.2 [2]

BASES

SURVEILLANCE REQUIREMENTS

[1] and by attempting to select a control rod not in compliance with the prescribed sequence and verifying a selection error occurs

at ≤ 10% RTP [4]

[6] Insert from page B 3.3-53 and B 3.3-54

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

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

, and [2]

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

Insert SR 3.3.2.1.2 [1]

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. If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8. The ~~18~~ 24 [6] month Frequency is based on the actual trip setpoint methodology utilized for these channels. [1]

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)

The Note to SR 3.3.2.1.3 allows a THERMAL POWER reduction to ≤ 10% RTP in MODE 1 to perform the required Surveillances if the 92 day Frequency is not met per SR 3.0.2. [2]

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.

TSTF-205

Insert SR 3.3.2.1.7

SR 3.3.2.1.~~6~~ 7 [6]

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.

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

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

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.

SR 3.3.2.2.2

2

TSTF-205

Insert SR 3.3.2.2.2

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 ~~reliability analysis (Ref. 2)~~ operating experience. 1

SR 3.3.2.2.~~3~~ 2 1

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

All changes are [1] unless indicated 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 narrow range instruments are calibrated to be accurate at the normal operating pressure and temperature.

3. ~~Suppression Pool~~ Water Level

Torus

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~~ bottom to the top of the ~~pool~~. Two wide range ~~suppression pool~~ water level signals are transmitted from separate differential pressure transmitters to two control room indicators and ~~are~~ also continuously ~~recorded~~ displayed on two recorders in the control room. These ~~recorders~~ instruments 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

4. Drywell Pressure

Type A and

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

The wide range drywell pressure

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

Insert LCO 4

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 the drywell pressure.

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

~~Primary containment area~~ radiation ~~(high range)~~ is a Category I 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. ~~For~~

(continued)

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

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 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 1 1

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1.1 (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.

1 1

SR 3.3.4.1.2 3 Insert SR 3.3.4.1.3 TSTF-205 C

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

2

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.

3 3

1 1 SR 3.3.4.1.3 2 1

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

move to page B 3.3-99 as indicated 1

(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.

C

2

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 8 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 (> 1 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.

2

Insert ASA-1

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.

Some Functions (i.e, Functions 1.c, 1.d, 2.c, 2.d, 4.d, 4.e, 5.d, and 5.e) have both an upper and lower analytic limit that must be evaluated. The Allowable Values and trip setpoints are derived from both an upper and lower analytic limit using the methodology describe above. Due to the upper and lower analytic limits, Allowable Values of these Functions appear to incorporate a range. However, the upper and lower Allowable Values are unique, with each Allowable Value associated with one unique analytic limit and trip setpoint. C

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.

Insert SR 3.3.5.1.2

TSTF-205

SR 3.3.5.1.2 and SR 3.3.5.1.4 4

1

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

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

of 31 days for SR 3.3.5.1.2 is based on engineering judgement and the reliability of the equipment. The Frequency

The Frequency of 92 days for SR 3.3.5.1.4 is based on the reliability analyses of Reference ~~5~~. 4

(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

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 ~~F~~requency 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.

and SR 3.3.5.2.4 5

SR 3.3.5.2.2

5

The 31 day Frequency of SR 3.3.5.2.2 is based on the reliability of the components

1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. Insert SR 3.3.5.2.2 TSTF-205

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

SR 3.3.5.2.4

The 92 day Frequency of ~~92 days~~ is based on the reliability analysis of Reference 1. 5

~~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~~ 5

(continued)

TSTF -205

INSERT SR 3.3.5.2.2

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

1 Insert BKGD-6

Each channel inputs into 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 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. 1C

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 strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the RHR 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. 1C

4 Insert Function 1.c

1.c. Main Steam Line Pressure—Timer

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 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. 6). 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 transients but short enough as to prevent excessive RPV depressurization.

This Function isolates the Group 1 valves.

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.

TSTF-205

Insert SR 3.3.6.1.2

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 ~~entire~~ channel will perform the intended function.

2

2

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

e 6

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9

10

4

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analyses described in References ~~6~~ and ~~7~~. The

24 month

~~184 day~~ 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)~~.

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.

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

e 9 10 1

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 ~~an~~ 4 ~~18~~ 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. 4

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 ~~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 4

(continued)

BASES

6

SURVEILLANCE REQUIREMENTS

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

SR 3.3.6.1.7 (continued)

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

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.

4

(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.

SR 3.3.6.2.2

TSTF-205

Insert SR 3.3.6.2.2 2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ 2 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 References ~~5~~ 4 and ~~6~~ 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 -2a5

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.

Relief Valve

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

~~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.~~

1

TSTF-205 not shown

~~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).~~

~~SR 3.3.6.3.5~~

1

~~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.~~

SR 3.3.6.3.~~6~~ 1 — 1

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 1 unless otherwise indicated

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. 5

SR 3.3.7.1.2

TSTF-205 Insert SR 3.3.7.1.2

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

2 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 and 6~~ 4. 1

SR 3.3.7.1.3

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

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

(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

SURVEILLANCE REQUIREMENTS

SR 3.3.7.2.1 (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 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 rate. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the required channels of this LCO.

SR 3.3.7.2.2

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

The Frequency of 92 days is based on the reliability analysis of Reference 4.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.2.3 and SR 3.3.7.2.4

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. A Note to SR 3.3.7.2.3 states that radiation detectors are excluded from CHANNEL CALIBRATION since they are calibrated in accordance with SR 3.3.7.2.4.

The Frequency of SR 3.3.7.2.3 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift associated with the channel, except for the radiation detectors, in the setpoint analysis. The Frequency of SR 3.3.7.2.4 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift for the radiation detector in the setpoint analysis.

SR 3.3.7.2.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the mechanical vacuum pump breakers and isolation valve 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 or the isolation valve 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.

(continued)

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

SR 3.3.8.1.~~2~~ 1 4 and SR 3.3.8.1.3 4 B

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. Insert SR 3.3.8.1.1 1 A Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequenc~~y~~ies of ~~31 days~~ 18 months and 24 months ~~is~~ 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 ~~31 day~~ 18 month or 24 month interval, as applicable, is a rare event. 4 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. C

BASES

BACKGROUND (continued)

3 inservice

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 MG set exceeds predetermined limits of overvoltage, undervoltage, or underfrequency, a trip coil driven by this logic circuitry opens the circuit breaker, which removes the associated power supply from service.

1 or alternate power supply

(undervoltage release coil) within the circuit breaker 1

APPLICABLE SAFETY ANALYSES

1 RPS Components

The RPS electric power monitoring is necessary to meet the assumptions of the safety analyses by ensuring that the equipment powered from the RPS buses can perform its intended function. RPS electric power monitoring provides protection to the RPS ~~and other systems that receive power from the RPS buses~~, by acting to disconnect the RPS from the power supply under specified conditions that could damage the RPS ~~bus powered~~ equipment. 3 3 1 1 bus 3

3 RPS electric power monitoring satisfies Criterion 3 of ~~the NRC Policy Statement~~. 1

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 inservice electric power monitoring assembly's trip logic setpoints are required to be within the specified Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. 3 of the is

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.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.2.1

Insert SR 3.3.8.2.1 TSTF-205

A CHANNEL FUNCTIONAL TEST is performed on each overvoltage, undervoltage, and underfrequency channel to ensure that the 3 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 SR 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.

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 placed 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. C

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)

3.4.5 RCS Leakage Detection Instrumentation

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

a. Drywell floor drain sump monitoring system; and

b. Primary containment atmospheric particulate monitoring system.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain sump monitoring system inoperable.	-------------NOTE------------ LCO 3.0.4 is not applicable. ----------------------------- A.1 Restore drywell floor drain sump monitoring system to OPERABLE status.	 30 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Primary containment atmospheric particulate monitoring system inoperable.	------------NOTE------------ LCO 3.0.4 is not applicable. -----------------------------	
	B.1 Analyze grab samples of primary containment atmosphere.	Once per 12 hours
	<u>AND</u>	
	B.2 Restore primary containment atmospheric particulate monitoring system to OPERABLE status.	30 days
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	C.2 Be in MODE 4.	36 hours
D. All required leakage detection systems inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the other required leakage detection instrumentation is OPERABLE.

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Perform a CHANNEL CHECK of the the primary containment atmospheric particulate monitoring system.	12 hours
SR 3.4.5.2	Perform a CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation.	31 days
SR 3.4.5.3	Perform a CHANNEL CALIBRATION of required leakage detection instrumentation.	24 months

C

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	---------NOTE-------- Only applicable if both RHR shutdown cooling subsystems are inoperable. --------------------- Verify reactor coolant circulating by an alternate method.	1 hour <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>		
	A.3	---------NOTE-------- Only applicable if both RHR shutdown cooling subsystems are inoperable. --------------------- Monitor reactor coolant temperature and pressure.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify each RHR shutdown cooling subsystem manual and power operated valve in the flow path, that is not locked, sealed or otherwise secured in position, is in the correct position or can be aligned to the correct position.	12 hours

BASES

BACKGROUND (continued)	Two drywell floor drain sump pumps take suction from the drywell floor drain sump and discharge to the Liquid Radioactive Waste System. When a high level is reached in the floor drain sump, a level switch actuates to start a floor drain sump pump when the pump discharge valves are open. A flow monitor in the discharge line of the drywell floor drain sump pumps provides a flow input to a flow integrator in the control room. The flow integrator is used to quantify the amount of sump input. The pumps can also be started from the control room. The primary containment atmospheric particulate monitoring system continuously monitors the primary containment atmosphere for airborne particulate radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The primary containment atmospheric particulate monitoring system is not capable of quantifying LEAKAGE rates, but satisfies the Regulatory Guide 1.45 (Ref. 2) recommended sensitivity of 1.0E-9 μCi/cc radioactivity for airborne particulates.
APPLICABLE SAFETY ANALYSES	A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 4 and 5). The drywell floor drain sump monitoring system is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits. The primary containment atmospheric particulate monitoring system provides indication of changes in leakage rates. A control room alarm, provided by the primary containment atmospheric particulate monitoring system, allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6). Therefore, these actions provide adequate response before a significant break in the RCPB can occur. RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

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

APPLICABILITY

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

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the primary containment atmospheric particulate monitoring system will provide indication of changes in leakage.

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1 and B.2

With the primary containment atmospheric particulate monitoring system inoperable, grab samples of the primary containment atmosphere must be taken and analyzed to provide periodic leakage information. Provided a sample is obtained

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of the required monitor.

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

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

C.1 and C.2

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

D.1

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

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 other required instrumentation (either the drywell floor drain sump monitoring system or the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

primary containment atmospheric particulate monitoring system, as applicable) is OPERABLE. 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. The 6 hour testing allowance is acceptable since it does not significantly reduce the probability of properly monitoring RCS leakage.

SR 3.4.5.1

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

SR 3.4.5.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

REFERENCES

1. UFSAR, Sections 3.1.3.6 and 3.1.6.4.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Section 5.2.5.7.
4. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.
5. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.
6. UFSAR, Section 5.2.5.5.

L

ITS 3.4.1

3.6 - LIMITING CONDITIONS FOR OPERATION

C. Recirculation Pumps

LCO 3.4.1 Recirculation pump speed shall be maintained within:

SR 3.4.1.1

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.

L.4

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

Page 3 of 3

ADMINISTRATIVE

A.4 (cont'd) instrumentation. Reference to the Trip Setpoints has been eliminated in the 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 Not used. 1c

M.2 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.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 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 limits 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 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. B

LA.2 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 (cont'd) 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.

LA.3 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.

"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, "RCS 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.

RELOCATED SPECIFICATIONS

None

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
≤1115 psig
≤1115 psig
≤1135 psig
≤1135 psig
≤1135 psig[a]

(see ITS 3.3.6.3)

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

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.

(see ITS 3.3.6.3)

2. Deleted.

add proposed SRs 3.4.3.2 and 3.4.3.3

A.4

LA.3

~~a Target Rock combination safety/relief valve.~~

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 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 do 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, endorsed by the NRC for Quad Cities 1 and 2. Compliance with 10 CFR 50.55a, and as a result the IST Program and implementing procedures, is required by the Quad Cities 1 and 2 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.

LA.3 The detail in CTS 3.6.E LCO footnote (b) that the lowest setting safety valve is a "Target Rock" combination safety/relief valve and the detail in CTS 3.6.F LCO footnote (a) that one of the relief valves is a "Target Rock" combination safety/relief valve are proposed to be relocated to the Bases. Proposed LCO 3.4.3 will continue to require 9 safety valves and 5 relief valves to be OPERABLE. In addition, the Bases provides sufficient description of the safety valves, relief valves, and the safety/relief valve. This detail is not necessary to ensure lift 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.

DISCUSSION OF CHANGES
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE

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

A.2 CTS 4.6.H.1 requires sampling the primary containment atmosphere particulate radioactivity at least once every 12 hours. Quad Cities 1 and 2 currently continuously monitor the primary containment atmosphere to satisfy this requirement. The CTS 4.6.H.1 requirement has been moved to proposed SR 3.4.5.1 requiring a CHANNEL CHECK of the primary comtainment atmospheric particulate monitoring system. This change is considered administrative since current practice is consistent with the CHANNEL CHECK definition. C

A.3 The CTS 3.6.G Action 2 requirement to establish an alternate method of determining drywell floor drain sump flow rates within 8 hours has been deleted from the proposed ITS 3.4.5 ACTIONS. CTS 4.6.H.2 (proposed ITS SR 3.4.4.1) already requires the determination of drywell floor drain sump flow rate at this specified frequency and 3.6.H (proposed ITS 3.4.5) provides the appropriate ACTIONS for conditions when LEAKAGE cannot be determined. In addition, proposed SR 3.4.4.1 Bases address acceptable methods to determine LEAKAGE. As such, the deleted information is unnecessary and this change is considered administrative. Any technical changes to CTS 4.6.H.2 will be addressed in the Discussion of Changes for ITS 3.4.5.

A.4 Currently, no Actions are provided in CTS 3.6.G if all required RCS leakage detection systems are inoperable. Therefore, CTS 3.0.C must be entered. The revised presentation of ACTIONS is proposed to explicitly identify that LCO 3.0.3 (CTS 3.0.C) is required to be entered if all required RCS leakage detection systems are inoperable. Therefore ITS 3.4.5 ACTION D has been added to be consistent with the current requirements and is considered a presentation preference. Therefore, this change is administrative.

A.5 The requirement in CTS 4.6.G.1 to perform the leakage determinations of CTS 4.6.H has been deleted since it duplicates the requirement of CTS 4.6.H.2 (proposed ITS SR 3.4.4.1). Therefore, this change is considered administrative.

ITS 3.4.8

3.6 - LIMITING CONDITIONS FOR OPERATION

P. Residual Heat Removal - COLD SHUTDOWN

LCO 3.4.8 Two shutdown cooling mode subsystems of the residual heat removal (RHR) system shall be OPERABLE[(a)(b)] A.2 and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem shall be capable of circulating reactor coolant[(b)] with each subsystem consisting of at least:

1. One OPERABLE RHR pump, and
2. One OPERABLE RHR heat exchanger. LA.1

APPLICABILITY:

OPERATIONAL MODE 4. add proposed LCO 3.4.8 Note L.1

ACTION: add proposed ACTION Note A.4

ACTION A

1. With less than the above required RHR shutdown cooling mode subsystems OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode subsystem.

4.6 - SURVEILLANCE REQUIREMENTS

P. Residual Heat Removal - COLD SHUTDOWN

SR 3.4.8.1 At least one shutdown cooling mode subsystem of the residual heat removal system, recirculation pump or alternate method shall be verified to be capable of circulating reactor coolant at least once per 12 hours. LA.2 M.1 moved to Required Action A.2

Required Action A.2

LA.1 A.2

a 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.

LCO 3.4.8 Note 1 b The RHR shutdown cooling subsystem(s) is not required to be OPERABLE and may be removed from operation during hydrostatic testing.

ADMINISTRATIVE

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

A.2 CTS 3.6.P requires two RHR shutdown cooling systems to be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem capable of circulating reactor coolant. CTS 3.6.P footnote (a) clarifies that an OPERABLE shutdown cooling subsystem is one than can be manually aligned in the shutdown cooling mode for removal of decay heat. This latter requirement (i.e., the subsystem can be manually aligned in the shutdown cooling mode for decay heat removal) also ensures proper circulation of reactor coolant since water obtained from the reactor vessel (via a reactor recirculation suction line) is pumped through an RHR heat exchanger (or heat exchange bypass valve) and routed to the reactor vessel via a recirculation discharge line. This will ensure the capability to circulate reactor coolant. Therefore, the detail in CTS 3.6.P that an RHR shutdown cooling subsystem must be capable of circulating reactor coolant is duplicative to the requirement in CTS 3.6.P footnote (a) and the definition of OPERABLE-OPERABILITY and has been deleted. Proposed LCO 3.4.8 will simply require both RHR shutdown cooling subsystems to be OPERABLE. Since this change does not result in any technical changes, the removal of the redundant requirement is considered administrative.

The content of CTS 3.6.P footnote (a) has been incorporated in proposed SR 3.4.8.1 (CTS 4.6.P) and will require the verification that each RHR shutdown cooling subsystem manual and power operated valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position, or can be aligned to the correct position. This proposed Surveillance is consistent with current requirements in CTS 4.6.P and is consistent with the format used in other Specifications in the CTS/ITS (e.g., ITS 3.1.7 for the Standby Liquid Control System) which require a system to be in standby. CTS 3/4.6.P does not require continuous reactor coolant circulation by a recirculation pump, an RHR shutdown cooling subsystem, or an alternate method as required by NUREG-0123 and NUREG-1433, Rev. 1. This deviation from the Standard Technical Specifications was approved by the NRC in the SER for Amendments 162 (Unit 1) and 158 (Unit 2) from John F. Stang (NRC) to D.L. Farrer (ComEd), dated September 21, 1995, since the RHR System and the RHR Service Water System flow at Quad Cities 1 and 2 can not be throttled sufficiently to maintain constant

ADMINISTRATIVE

A.2 (cont'd) temperature. Therefore, the system must be used continuously or intermittently to control temperature and be within the required cooldown/heatup rates. In MODE 4 RHR shutdown cooling subsystem is turned on and off as necessary to control temperature and to ensure the required cooldown or heatup rates of CTS 3/4.6.K (ITS 3.4.9) are not exceeded. The requirement to have a recirculation pump, RHR shutdown cooling subsystem or alternate method in operation is governed by plant procedures, except when both RHR shutdown cooling subsystems are inoperable. In this situation, circulation is required per CTS 3.6.P Action 2 (proposed ITS 3.4.8 ACTION A). Since this change simply represents a change in presentation, this portion of the change is also administrative.

A.3 Not used.

A.4 The proposed ACTION Note, "Separate Condition entry is allowed for each RHR shutdown cooling subsystem" has been added to CTS 3.6.P Actions (ITS 3.4.8 ACTIONS Note) and provides more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," this Note provides direction consistent with the intent of the existing Actions for inoperable RHR shutdown cooling subsystems.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 4.6.P requires the verification of the capability to circulate reactor coolant via an RHR shutdown cooling subsystem, a recirculation pump, or an alternate method every 12 hours. As indicated in Discussion of Change A.2 above, the proposed Surveillance has been written to ensure the requirements of the LCO are met. Since there are no actual requirements for recirculation pumps or an alternate method in the LCO, the option to verify the alternate method has been included as ITS 3.4.8 Required Action A.2 (refer to Discussion of Change LA.2 below for relocation of the recirculation pump as an alternate method of decay heat removal to the Bases) for the condition when both RHR shutdown cooling subsystems are inoperable since this is the only time an alternate method is required to be in operation. The requirement in ITS 3.4.8 Required Action A.2 is to verify reactor coolant circulating by an alternate method once per 12 hours. Since the new requirement requires the alternate method to be in operation (i.e., actually circulating reactor coolant) in lieu of the current requirement that the alternate method be capable of circulating reactor coolant, this change is considered more restrictive on plant operation. This will provide assurance of continued temperature monitoring capability.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details in CTS 3.6.P.1 and CTS 3.6.P.2 of what constitutes an OPERABLE RHR shutdown subsystem and the detail in CTS 3.6.P footnote (a) describing how the system can be aligned "remote or local" are proposed to be relocated to the Bases. The Bases will indicate that an OPERABLE RHR shutdown cooling subsystem consists of an OPERABLE pump, heat exchanger, service water providing cooling to the heat exchanger, and the associated piping and valves. Proposed SR 3.4.8.1 indicates that a system can be considered OPERABLE if it can be aligned to the correct position. The details for subsystem OPERABILITY are not necessary in ITS 3.4.8. The definition of OPERABILITY suffices. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 The detail of an acceptable alternate method in CTS 4.6.P and CTS 3.6.P Action 2 for circulating reactor coolant (i.e., a recirculation pump) is proposed to be relocated to the Bases. This detail is not necessary for assuring circulation of the reactor coolant when necessary. Proposed ITS 3.4.8 Required Action A.2 requires verification of reactor coolant circulation by an alternate method. 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.

"Specific"

L.1 CTS 3.6.P does not allow one RHR shutdown cooling subsystem to be inoperable for performance of Surveillances. An allowance has been added (proposed LCO 3.4.8 Note) which allows one RHR shutdown cooling subsystem to be inoperable up to 2 hours to perform Surveillances. This change is necessary since tests are required that necessitate placing the RHR Shutdown Cooling System in an inoperable status. This is acceptable because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR shutdown cooling subsystems or other operations requiring RHR flow interruptions and loss of redundancy. This change is consistent with BWR ISTS, NUREG-1433, Rev. 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.2 With no RHR shutdown cooling subsystem OPERABLE, CTS 3.6.P Action 2 requires the plant to immediately initiate corrective action to return at least one subsystem to OPERABLE status as soon as possible. In addition, the same action requires reactor coolant circulation to be established with a recirculation pump or by an alternate method within 1 hour and to monitor reactor coolant temperature and pressure at least once per hour. During MODE 4 operations, the core decay heat may be low enough such that alternative methods of decay heat removal are sufficient to remove core decay heat to maintain or reduce temperature. CTS 3.6.P Action 1 (ITS 3.4.8 Required Action A.1) requires the verification that an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem within one hour and once per 24 hours thereafter. In this MODE, it is acceptable to take both RHR subsystems out of service to perform required maintenance operations since the Technical Specifications provide sufficient requirements to ensure decay heat removal methods are available. Therefore, the requirement in CTS 3.6.P Action 2 to immediately initiate corrective action to return at least one subsystem to OPERABLE status as soon as possible has been deleted. This allowance has been approved at both Dresden and LaSalle and is consistent with BWR ISTS, NUREG-1433, Rev. 1.

RELOCATED SPECIFICATIONS

None

A.1

ITS 3.4.9

A.5

add proposed SR 3.4.9.7 Note

3.6 - LIMITING CONDITIONS FOR OPERATION

3. Nuclear Heatup and Cooldown:

SR 3.4.9.2 — a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Region as shown on Figure 3.6.K-3, and

SR 3.4.9.1 — b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour.

SR 3.4.9.5, SR 3.4.9.6, SR 3.4.9.7 — 4. The reactor vessel flange and head flange temperature ≥83 °F when reactor vessel head bolting studs are under tension.

APPLICABILITY:

At all times.

ACTION:

add proposed Notes for Condition A and C — A.5, A.2

With any of the above limits exceeded,

ACTION A, ACTION 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.9

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.2, L.1

ACTION B — 3. Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

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:

SR 3.4.9.7 — 1) ≤113°F, at least once per 12 hours.

SR 3.4.9.6 — 2) ≤93°F, at least once per 30 minutes.

A.6

SR 3.4.9.5 — b. ~~Within 30 minutes prior to and~~ at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

add proposed SR 3.4.9.6 Note

ADMINISTRATIVE (continued)

A.5 CTS 4.6.K.4.a requires periodic verification that reactor vessel and head flange temperatures are ≥ 83°F. The Frequency of this verification change is based on reactor coolant system temperature. Notes have been provided in proposed SR 3.4.9.6 and 3.4.9.7 to clarify the current intent in CTS 4.6.K.4.a of allowing entry into the applicable conditions (i.e., ≤ 113°F and ≤ 93°F) without having performed these Surveillance Requirements. Since this requirement is currently only performed during the specified conditions (i.e., when ≤ 113°F and ≤ 93°F), these changes (the addition of the two Notes) are considered administrative.

A.6 The CTS 4.6.K.4.b requirement to verify the reactor vessel and head flange temperatures within 30 minutes prior to tensioning of the head bolting studs has been deleted. This requirement is duplicative of CTS 4.0.A and proposed SR 3.0.1, which require the Surveillance to be current when in the applicable Mode or condition. CTS 4.0.C and proposed SR 3.0.1 also state that failure to meet the Surveillance constitutes failure to meet the LCO, which would then require the ACTIONS of the LCO to be taken. CTS 3.6.K Action 1 (ITS 3.4.9 ACTION C) requires action to be taken to restore the limit. Therefore, this effectively ensures that the Applicability of this SR (as stated in the Note to the SR) is not entered when CTS 4.6.K.4.b (proposed SR 3.4.9.5) is not current. Therefore, this change is considered administrative.

A.7 The CTS 3.6.D requirements have been combined into the RCS P/T Limits Specification, with the words "and the recirculation pump starting temperature requirements" added to the ITS 3.4.9 LCO statement. The actual description of the requirements and the limits are found in proposed SR 3.4.9.3 and SR 3.4.9.4. As such, this change is administrative.

A.8 Thermal stresses on vessel components are dependent upon the temperature difference between the idle loop coolant and the RPV coolant. CTS 3.6.D.1 and 3.6.D.2 (proposed SR 3.4.9.4) ensure the temperature difference between the idle loop and the RPV coolant is acceptable. The CTS 3.6.D.2 requirement to monitor the temperature difference between an idle loop and an operating loop is unnecessary and has been deleted since it is redundant to the loop-to-coolant requirement of CTS 3.6.D.1 (proposed SR 3.4.9.4). However, the loop-to-coolant temperature check may use the operating loop temperature as representative of "coolant temperature."

A.9 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 C

ADMINISTRATIVE

A.9 (cont'd) 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 The CTS 3.6.D Action required to be taken when a recirculation pump is started without having met the temperature requirements has been changed. Currently, the CTS 3.6.D Action only states to suspend the startup of a recirculation loop. This however, does not provide an action if the loop is already operating. Proposed ITS 3.4.9 adds ACTIONS A, B, and C which, in this condition, would require an engineering evaluation to be performed to ensure continued operation is acceptable. This is an additional restriction on plant operation necessary to ensure the RCS is acceptable for continued operation.

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 all times 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 of CTS 3.6.D (and its associated Action) and CTS 4.6.D relating to operational limits (loop flow and pump speed) 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 and pump speed are considered operational limits since they are not directly related to the ability of the system to perform its safety analysis functions. The flow rate and pump speed are limited only to restrict reactor vessel internals vibration to within acceptable limits during restart of the second pump. These requirements are

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) 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.

LA.2 The detail 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.3 CTS 3.6.D requires that an idle recirculation loop not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is ≤ 145 F. The detail on where the temperature is monitored (e.g., space or line) to evaluate the temperature difference is proposed to be relocated to the Bases. The requirement in proposed ITS SR 3.4.9.3 to verify the difference between the bottom head coolant temperature and the reactor pressure vessel coolant temperature is sufficient to ensure that the differential temperature is met prior to a startup of a recirculation pump. The Bases for SR 3.4.9.3 indicates an acceptable method for evaluating the limit. 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.

"Specific"

L.1 CTS 3.6.K Action 2 specifies 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

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) 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.

L.2 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-3) 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.

RELOCATED SPECIFICATIONS

None

<CTS>

<3.6.6 Act 1>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ~~Required~~ primary containment atmospheric particulate monitoring system inoperable. 7 3	------------NOTE------------ LCO 3.0.4 is not applicable. ---------------------------- 5 2	
	B.1 Analyze grab samples of primary containment atmosphere.	Once per 12 hours
	AND	
	B.2 Restore ~~required~~ primary containment atmospheric particulate monitoring system to OPERABLE status. 7 3	30 days 2
~~C. Primary containment air cooler condensate flow rate monitoring system inoperable.~~	~~C.1 ---------NOTE--------- Not applicable when required primary containment atmospheric monitoring system is inoperable. --------------------- Perform SR 3.4.6.1.~~	~~Once per 8 hours~~ 4

(continued)

<CTS>

<DOC L.3>

SURVEILLANCE REQUIREMENTS Insert SR NOTE [6]

	SURVEILLANCE	FREQUENCY
<4.6.H.1> [1]	SR 3.4.~~6~~5.1 Perform a CHANNEL CHECK of ~~required~~ the primary containment atmospheric particulate monitoring system. [7] [3]	12 hours
<DOC M.1> [1]	SR 3.4.~~6~~5.2 Perform a CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation.	31 days
<4.6.G.2> [1]	SR 3.4.~~6~~5.3 Perform a CHANNEL CALIBRATION of required leakage detection instrumentation.	~~[18]~~ 24 months [2]

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

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

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

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

4. The bracketed requirement/information has been deleted since it is not applicable to Quad Cities 1 and 2. The following requirements have been renumbered, where applicable, to reflect this deletion.

5. TSTF-60 moved the Notes in ISTS ACTIONS A and B to the beginning of the ACTIONS Table. This implies the Note is applicable to all the ACTIONS. However, the Note "LCO 3.0.4 in not applicable" is not applicable to proposed ISTS ACTIONS C and D, which require a plant shutdown. Therefore, to avoid confusion, TSTF-60 has not been incorporated.

6. A Note has been added to allow a channel to be inoperable for up to 6 hours solely for performance of required Surveillances, provided the other Leakage Detection System instrumentation is OPERABLE. This Note is similar to other Notes in the ITS, which allow channels that provide automatic actions to be inoperable for up to 6 hours. This instrumentation only provides indication, and the 6 hour allowance is not allowed to be used unless the other channel is OPERABLE. This change has been previously approved at Georgia Power Company's Plant Hatch Units 1 and 2, in Amendments 185 and 125, respectively, and in the ITS conversion for Washington Public Power Supply System's WNP-2, Amendment 149.

7. The use of the word "required" has been deleted since each unit has only one channel of primary containment atmospheric particulate monitoring. C

<CTS>

- - - - - - Note - - -
Only applicable if both RHR shutdown cooling subsystems are inoperable.

ACTIONS (continued)

CONDITION		REQUIRED ACTION	COMPLETION TIME
<3.6.P Act 1> <3.6.P Act 2> ~~B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.~~	~~B.1~~ A.2	Verify reactor coolant circulating by an alternate method.	1 hour ~~from discovery of no reactor coolant circulation~~ AND Once per 12 hours thereafter [2]
	AND		
[A.3]	~~B.2~~	Monitor reactor coolant temperature and pressure. [5]	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.6.P> SR 3.4.~~9~~.1 [1] 8	Verify ~~one~~ each RHR shutdown cooling subsystem ~~or recirculation pump is operating~~ manual and power operated valve in the flow path that is not locked, sealed or other wise secured in position, is in the correct position or can be aligned to the correct position.	12 hours [2]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.8 - RHR SHUTDOWN COOLING SYSTEM — COLD SHUTDOWN

1. 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."

2. The requirement in ISTS 3.4.9 (ITS 3.4.8) to maintain a RHR shutdown cooling subsystem or recirculation pump in operation has been deleted. This deviation from the Standard Technical Specifications was approved by the NRC in the SER for Amendments 162 (Unit 1) and 158 (Unit 2) from John F. Stang (NRC) to D.L. Farrar (ComEd), dated September 21, 1995. As a result, the LCO, LCO Note 1, ACTIONS and Surveillances have been revised to reflect current allowances (refer to Discussion of Changes for further discussion). Since ISTS 3.4.9 Note 1 has been deleted, the changes approved in TSTF-153 are not shown.

3. Note 1 has been added to ISTS 3.4.9 (ITS 3.4.8) in order allow the performance of the hydrostatic test with both RHR shutdown cooling subsystems inoperable. This allowance in ITS 3.4.8 is necessary since CTS 3.12.C (ISTS 3.10.1, "Inservice Leak and Hydrostatic Testing Operation") has been deleted in accordance with the Technical Specifications Change Request submitted to the NRC for approval per ComEd letter SVP-99-193, dated November 12, 1999. Since CTS 3.12.C (ISTS 3.10.1) allowed the suspension of the requirements in CTS 3.6.P (ISTS 3.4.9) to allow the performance of inservice leak or hydrostatic test, this allowance will be needed in ITS 3.4.8. The RHR Shutdown Cooling System is inoperable during hydrostatic testing since the system is not capable of circulating reactor coolant. The RHR Shutdown Cooling System is automatically isolated above the RHR cut-in permissive pressure. This isolation is necessary since the RHR Shutdown Cooling System is not designed to operate at the Reactor Coolant System pressure achieved during hydrostatic testing. This proposed Note is consistent with the ISTS 3.10.1 allowance to suspend the requirements of the RHR Shutdown Cooling System-Cold Shutdown LCO during hydrostatic testing.

4. Editorial change made to be consistent with the LCO requirements and with a similar Note in ISTS 3.4.8. B

5. This change was made to be consistent with the current licensing basis and NUREG-1434, ISTS 3.4.10. C

BASES

BACKGROUND (continued)

~~If the sump fills to the high level setpoint before the timer ends, an alarm sounds in the control room, indicating a LEAKAGE rate into the sump in excess of a preset limit.~~ 2

Insert BCKGD 1

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

Insert BCKGD 2

2

The primary containment ~~air~~ atmospheric particulate monitoring system~~s~~ continuously monitor~~s~~ the primary containment atmosphere for airborne particulate ~~and gaseous~~ radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The primary containment atmospheric particulate ~~and gaseous~~ radioactivity monitoring system~~s~~ ~~are~~ not capable of quantifying LEAKAGE rates, ~~but are sensitive enough to indicate increased LEAKAGE rates of 1 gpm within 1 hour. Larger changes in LEAKAGE rates are detected in proportionally shorter times (Ref. 3).~~ 2

satisfies the Regulatory Guide 1.45 (Ref. 2) recommended sensitivity of 1.0 E-9 μCi/cc radioactivity for airborne particulates

~~Condensate from four of the six primary containment coolers is routed to the primary containment floor drain sump and is monitored by a flow transmitter that provides indication and alarms in the control room. This primary containment air cooler condensate flow rate monitoring system serves as an added indicator, but not quantifier, of RCS unidentified LEAKAGE.~~ 3

2

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 4 and 5). ~~Each of the leakage detection systems inside the drywell~~ The drywell floor drain sump monitoring system is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits~~, and providing appropriate alarm of excess LEAKAGE in the control room~~.

The primary containment atmospheric particulate monitoring system provides indication of changes in leakage rates. 2

2

A control room alarm, provided by the primary containment atmospheric particulate monitoring system, allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6). Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

the system 2 C 5 3

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

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

is methods are 2 C 1 3

[C.1

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

(continued)

1

Insert SR

BASES (continued)

1

SURVEILLANCE REQUIREMENTS

5

SR 3.4.6.1

particulate

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.

5 1

SR 3.4.6.2

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

2 2 TSTF-205

Insert SR 3.4.5.2

5 1

SR 3.4.6.3

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

5 24

2

UFSAR, Sections 3.1.3.6 and 3.1.6.4

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section [5.2.7.2.1]. 5.2.5.7
4. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.
5. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.
6. FSAR, Section [5.2.7.5.2]. 5.2.5.5

2 U 5 4 5

1 INSERT SR

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 other required instrumentation (either the drywell floor drain sump monitoring system or the primary containment atmospheric particulate monitoring system, as applicable) is OPERABLE. 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. The 6 hour testing allowance is acceptable since it does not significantly reduce the probability of properly monitoring RCS leakage.

TSTF -205 INSERT SR 3.4.5.2

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

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.

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.

C

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 <u>AND</u> ------NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass --------------- 12 months

C

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (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. UFSAR, Section 6.2.1.2.4.1.

BASES

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 one second. The air operated butterfly valve vacuum breakers are assumed to open concurrent with the mechanical vacuum breakers and be full open in one second (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. Four cases were considered in the safety analyses to determine the adequacy of the external vacuum breakers: a. A small steam line break loss of coolant accident followed by actuation of one drywell and suppression pool spray loop; b. An intermediate steam line break loss of coolant accident followed by actuation of one drywell and suppression pool spray loop;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. A postulated DBA assuming Emergency Core Cooling Systems (ECCS) runout flow with a thermal mixing efficiency of 60%; and

d. A postulated DBA followed by actuation of one drywell and suppression pool spray loop.

The results of these four 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.

(continued)

BASES

APPLICABILITY (continued)

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.

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

(continued)

ACTIONS

C.1 (continued)

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.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.1 (continued)

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.

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 specify

(continued)

BASES

ACTIONS

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

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 opening share the inner door or the outer door, i.e., the access openings have a common inner or outer door. The intent is to not breach the

(continued)

BASES

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)

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 15.7.2.
3. Technical Requirements Manual.

BASES

ACTIONS (continued)

C.1, C.2.1, C.2.2, and C.2.3

During movement of irradiated fuel assemblies, in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should immediately be placed in operation. This action 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

(continued)

ACTIONS

D.1 (continued)

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.

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)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating (from the control room using the manual initiation switch) each SGT subsystem for $\geq$ 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 $\geq$ 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.

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. 6). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

This SR verifies that each SGT subsystem starts on receipt of an actual or simulated initiation signal. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. 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. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.9.1.
2. UFSAR, Section 6.5.1.1.
3. UFSAR, Section 15.6.2.
4. UFSAR, Section 15.6.5.
5. UFSAR, Section 15.7.2.
6. Regulatory Guide 1.52, Rev. 2.

A.1

ITS 3.6.1.1

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 residual heat removal 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 3.6.2.2>

4.7 - SURVEILLANCE REQUIREMENTS

3. Deleted.

4. Deleted.

5. At least once per 18 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 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

SR 3.6.1.1.2

24 LD.1

L.3

L.2

C

24

12

LD.1

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 Quad Cities 1 and 2 refuel cycle from 18 months to 24 months. The proposed change will allow the normal Surveillance to extend the Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.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

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) passive design of the 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 Quad Cities Special Report No. 4 submitted to A. Giambusso (NRC) from L. D. Butterfield (Commonwealth Edison) on October 23, 1972. 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

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.

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 ≥0.10 inches of vacuum water gauge.

2. Verifying at least once per 31 days that:

   L.2

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

   add proposed Required Action A.1

   Required Action A.2 and SR 3.6.4.2.1

   M.2

   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.

   A.2

   and 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.

See ITS 3.6.4.1

Required Action A2 Note * When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

SR 3.6.4.2.1 Note 1 a Valves and blind flanges in high-radiation areas may be verified by use of administrative controls. Normally

SR 3.6.4.2.1 Note 2 ~~locked or sealed-closed~~ penetrations may be opened intermittently under administrative control. L.1

DISCUSSION OF CHANGES
ITS: 3.6.4.2 - SECONDARY CONTAINMENT ISOLATION VALVES (SCIVs)

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 Quad Cities 1 and 2 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 the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2) to a

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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