UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 15, 2000

Template
NRR-058

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, Second Floor
Lycoming, NY 13093

SUBJECT: CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 2 - AMENDMENT NO. 91 TO FACILITY OPERATING LICENSE NO. NPF-69 (TAC NO. MA3822)

Dear Mr. Mueller:

The Commission has issued the enclosed Amendment No. 91 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit 2 (NMP2). The amendment converts the current Technical Specifications (CTS) for NMP2 to the improved Technical Specifications (ITS). The ITS are based on NUREG-1433 and NUREG-1434 "Standard Technical Specifications for General Electric Plants, BWR/4 and BWR/6," Revision 1, dated April 1995, and on guidance provided in the Commission's Final Policy Statement, "NRC Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," published on July 22, 1993 (58 FR 39132), and in 10 CFR 50.36, Technical Specifications," as amended July 19, 1995 (60 FR 36953). The overall objective of the proposed amendment was to rewrite, reformat, and streamline the CTS to improve safety and the understanding of the Bases underlying the Technical Specifications.

The enclosed amendment is based on the staff's review of your application dated October 16, 1998 (NMP2L 1830), as supplemented by letters dated December 30, 1998 (NMP2L 1844), May 10 (NMP2L 1866), June 15 (NMP2L 1872), July 30 (NMP2L 1881), August 2 (NMP2L 1883), August 11 (NMP2L 1885), August 16 (NMP2L 1886), August 19 (NMP2L 1888), August 27 (NMP2L 1893), September 10, (NMP2L 1896), September 30, 1999 (NMP2L 1900), December 14, 1999 (NMP2L 1917), January 6, 2000 (NMP2L 1923), January 17, 2000 (NMP2L 1925), January 21, 2000 (NMP2L 1928), January 31, 2000 (NMP2L 1929), and February 11, 2000 (NMP2L 1936). The staff issued requests for additional information (RAIs) dated March 26, May 10, May 18, June 16, September 21, October 16, and November 10, 1999. Summaries of an NRC staff meeting with the licensee regarding the conversion was issued on December 8, 1999.

The draft Safety Evaluation (SE) for the ITS conversion was sent to you by letter dated December 13, 1999, for your review to verify the accuracy of the draft SE. You responded with comments on the draft SE by letter dated January 17, 2000 (NMP2L 1917). You submitted the ITS for NMP2 and certified their correctness in that you provided the affirmation in your letter of January 31, 2000 (NMP2 1929), that the statements made in that letter, and its attachments including the ITS, are true. The comments you provided were reviewed and incorporated in the enclosed final SE for the amendment as appropriate. The draft SE was also revised based on the staff's review of the draft SE after it was issued.

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Included with the amendment request were two proposed license conditions to the license of NMP2. You submitted proposed license conditions in your letter of January 17, 2000, regarding (1) the relocation of CTS requirements into licensee-controlled documents as part of the implementation of the ITS and (2) the schedule for the first performance of new and revised surveillance requirements (SRs) for the ITS. These license conditions are part of the implementation of the ITS and ensure enforceability of commitments that the staff relied upon in approving this amendment. Any changes to these license conditions, including the implementation date for the ITS conversion, must be submitted as a 10 CFR 50.90 license amendment request and must be approved by the staff.

The ITS will become effective upon the date of implementation, but no later than August 31, 2000. Until the implementation of the ITS is completed, the CTS shall remain in effect and the unit will be operated in accordance with the CTS. If there is an amendment to the Technical Specifications before the implementation of the ITS is completed, separate amendments to both the CTS and the ITS will be required. You are requested to submit a letter stating that the ITS are implemented on the unit within 14 days of the date of implementation.

The review and issuance of this amendment has been a significant challenge to the NRC staff in that this ITS conversion contained more beyond scope issues than typical for previously issued amendments for the conversion of the CTS to the ITS. Also, the staff schedule was challenged due to your staff's reluctance to attend the important meetings of October 20 and 21, 1999, to resolve open issues. Complications were also encountered when you recently requested proposed license amendments in the CTS format and did not recognize that one was not timely and all should also have been sent in the proposed ITS format.

A copy of the related Safety Evaluation is enclosed. Also enclosed is the Notice of Issuance which has been forwarded to the Office of the Federal Register for publication.

Contact me at 301-415-1441 (or gsv@nrc.gov on the internet) if you have any questions regarding the amendment.

Sincerely,

Guy S. Vissing

Guy S. Vissing, Senior Project Manager, Section I
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures: 1. Amendment No. 91 to NPF-69
2. Safety Evaluation
3. Notice of Issuance

cc w/encls: See next page

Included with the amendment request were two proposed license conditions to the license of NMP2. You submitted proposed license conditions in your letter of January 17, 2000, regarding (1) the relocation of CTS requirements into licensee-controlled documents as part of the implementation of the ITS and (2) the schedule for the first performance of new and revised surveillance requirements (SRs) for the ITS. These license conditions are part of the implementation of the ITS and ensure enforceability of commitments that the staff relied upon in approving this amendment. Any changes to these license conditions, including the implementation date for the ITS conversion, must be submitted as a 10 CFR 50.90 license amendment request and must be approved by the staff.

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Sincerely,

Guy S. Vissing, Senior Project Manager, Section I
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures: 1. Amendment No. to NPF-69
2. Safety Evaluation
3. Notice of Issuance

cc w/encls: See next page

DOCUMENT NAME: G:\PDI-1\NMP2\AMDA3822.WPD *See previous concurrence.

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure **"E"** = Copy with attachment/enclosure **"N"** = No copy

OFFICE	PDI-1\PM		**PDI-1\LA**		**OGC***		**PDI\ASC**		BC\RTSB*		**PDI-1/PM**	
NAME	PTam		SLittle		**RHoefling**		MGamberoni		WBeckner		**GVissing**	
DATE	2/15/00		2/15/00		**2/15/2000**		2/15/00		**2/7/2000**		2/16/00	

Official Record Copy

Nine Mile Point Nuclear Station
Unit No. 2

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Note:

To receive a copy of the Technical Specifications, please call Guy Vissing at (301) 415-1441.

DATED: February 15, 2000

AMENDMENT NO. 91 TO FACILITY OPERATING LICENSE NO. NPF-69-NINE MILE POINT UNIT 2

File Center
PUBLIC
PDI Reading File
J. Zwolinski\S. Black
M. Gamberoni
S. Little
P. Tam
G. Vissing
OGC
G. Hill (2), T-5 C3
W. Beckner, 013H3
ACRS
PD plant-specific file
M. Oprendek, Region I

cc: Plant Service list

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 91
License No. NPF-69

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated October 16, 1998 (NMP2L 1830), as supplemented by letters dated December 30, 1998 (NMP2L 1844), May 10 (NMP2L 1866), June 15 (NMP2L 1872), July 30 (NMP2L 1881), August 2 (NMP2L 1883), August 11 (NMP2L 1885), August 16 (NMP2L 1886), August 19 (NMP2L 1888), August 27 (NMP2L 1893), September 10, (NMP2L 1896), September 30, 1999 (NMP2L 1900), December 14, 1999 (NMP2L 1923), January 6, 2000 (NMP2L 1923), January 17, 2000 (NMP2L 1925), January 21, 2000 (NMP2 1928), January 31, 2000 (NMP2L 1929), and February 11, 2000 (NMP2L 1936), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-69 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 91 are hereby incorporated into this license. Niagara Mohawk Power Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

In addition, the license is amended to add paragraphs 2.C.(10) and 2.C.(11) to Facility Operating License No. NPF-69 as follows: NMPC shall operate the facility in accordance with the Additional Conditions.

(10) The licensee is authorized by Amendment No. 91 to relocate certain Technical Specification requirements previously included in Appendix A to licensee-controlled documents, as described in Table R, Relocated Specifications and Removal of Details Matrix, attached to the NRC Staff's safety evaluation dated February 15, 2000, enclosed with the amendment. Implementation of Amendment No. 91 shall include the relocation of these requirements to the appropriate documents, which shall be completed no later than August 31, 2000. The relocations to the Updated Safety Analysis Report shall be reflected in updates completed in accordance with 10 CFR 50.71(e).

(11) The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 91 shall be as follows:

For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.

For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.

For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.

For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.

3. This license amendment is effective as of the date of its issuance and shall be implemented by August 31, 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Peter S. Tam for

Marsha Gamberoni, Acting Chief, Section I
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: 1. Pages 5 thru 8 of License No. NPF-69
2. Changes to Appendices A and D

Date of issuance: February 15, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 91

TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

1. Remove Facility Operating License No. NPF-69, pages 5 thru 8 and replace with pages 5 thru 8.

2. Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
All pages	All pages

(6) Initial Startup Test Program (Section 14, SER, SSERs 4 and 5)

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(7) Operation with Reduced Feedwater Temperature (Section 15.1. SSER 4)

Niagara Mohawk Power Corporation shall not operate the facility with reduced feedwater temperature for the purpose of extending the normal fuel cycle. The facility shall not be operated with a feedwater heating capacity less than that required to produce a feedwater temperature of 405°F at the rated steady-state conditions unless analyses supporting such operations are submitted by Niagara Mohawk Power Corporation and approved by the staff.

(8) Safety Parameter Display System (SPDS) (Section 18.2. SSERs 3 and 5)

Prior to startup following the first refueling outage, Niagara Mohawk Power Corporation shall have operational an SPDS that includes the revisions described in their letter of November 19, 1985. Before declaring the SPDS operational, the licensee shall complete testing adequate to ensure that no safety concerns exist regarding the operation of the Nine Mile Point Nuclear Station, Unit No. 2 SPDS.

(9) Detailed Control Room Design Review (Section 18.1. SSERs 5 and 6)

(a) Deleted per Amendment No. 24 (12-18-90)

(b) Prior to startup following the first refueling outage, Niagara Mohawk Power Corporation shall provide the results of the reevaluation of normally lit and nuisance alarms for NRC review in accordance with its August 21, 1986 letter.

(c) Prior to startup following the first refueling outage, Niagara Mohawk Power Corporation shall complete permanent zone banding of meters in accordance with its August 4, 1986 letter.

(10) Additional Condition 1

The licensee is authorized by Amendment No. 91 to relocate certain Technical Specification requirements previously included in Appendix A to licensee-controlled documents, as described in Table R, Relocated Specifications and Removal of Details Matrix, attached to the NRC Staff's safety evaluation dated February 15, 2000, enclosed with the amendment. Implementation of Amendment No. 91 shall include the relocation of these requirements to the appropriate documents, which shall be completed no later than August 31, 2000. The relocations to the Updated Safety Analysis Report shall be reflected in updates completed in accordance with 10 CFR 50.71(e).

(11) Additional Condition 2

The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 91 shall be as follows:

For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.

For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.

For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.

For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70.

i) An exemption from the criticality alarm requirements of 10 CFR Part 70.24 was granted in the Special Nuclear Materials License No. SNM-1895 dated November 27, 1985. This exemption is described in Section 9.1 of Supplement 4 to the SER. This previously granted exemption is continued in this operating license.

ii) Exemptions to certain requirements of Appendix J to 10 CFR Part 50 are described in Supplements 3, 4, and 5 to the SER. These include (a) (this item left intentionally blank); (b) an exemption from requirement of Option B of Appendix J, exempting main steam isolation valve measured leakage from the combined leakage rate limit of 0.6 La. (Section 6.2.6 of SSER 5)*; (c) an exemption from Option B of Appendix J, exempting the hydraulic control system for the reactor recirculation flow control valves from Type A and Type C leak testing (Section 6.2.6 of SSER 3); (d) an exemption from Option B of Appendix J, exempting Type C testing on traversing incore probe system shear valves. (Section 6.2.6 SSER 4)

* The parenthetical notation following the discussion of each exemption denotes the section of the Safety Evaluation Report (SER) and/or its supplementals wherein the safety evaluation of the exemption is discussed.

iii) An exemption to Appendix A to 10 CFR Part 50 exempting the Control Rod Drive (CRD) hydraulic lines to the reactor recirculation pump seal purge equipment from General Design Criterion (GDC) 55. The CRD hydraulic lines to the reactor recirculation pump seal purge equipment use two simple check valves for the isolation outside containment (one side). (Section 6.2.4, SSER 3)

iv) A schedular exemption to GDC 2, Appendix A to 10 CFR Part 50, until the first refueling outage, to demonstrate the adequacy of the downcomer design under the plant faulted condition. This exemption permits additional analysis and/or modifications, as necessary, to be completed by the end of the first refueling outage. (Section 6.2.1.7.4, SSER 3)

v) A schedular exemption to GDC 50, Appendix A to 10 CFR Part 50 to allow Niagara Mohawk Power Corporation until start-up following the "mini-outage," which is to occur within 12 months of commencing power operation (entering Operational Condition 1), to install redundant fuses in circuits that use transformers for redundant penetration protection in accordance with their letter of August 29, 1986 (NMP2L 0860). (Section 8.4.2, SSER 5)

vi) A schedular exemption to 10 CFR 50.55a(h) for the Neutron Monitoring System until completion of the first refueling outage to allow Niagara Mohawk Power Corporation to provide qualified isolation devices for Class IE/non-1E interfaces described in Niagara Mohawk Power Corporation's letters of June 23, 1987 (NMP2L 1057) and June 25, 1987 (NMP2L 1058). (Section 7.2.2.10, SSER 6).

For the schedular exemptions in iv), v), and vi), above, Niagara Mohawk Power Corporation, in accordance with its letter of October 31, 1986, shall certify that all systems, components, and modifications have been completed to meet the requirements of the regulations for which the exemptions have been granted and shall provide a summary description of actions taken to ensure that the regulations have been met. This certification and summary shall be provided 10 days prior to the expiration of each exemption period as described above.

The exemptions set forth in this Section 2.D are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security. These exemptions are hereby granted. The special circumstances regarding each exemption are identified in the referenced section of the Safety Evaluation Report and the supplements thereto. The exemptions in ii) through vi) are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans, including amendments made pursuant to the provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73. 21, entitled "Nine Mile Point Nuclear Station Physical Security Plan" with revisions submitted through June 9, 1994; "Nine Mile Point Nuclear Station Guard Training and Qualification Plan," with revisions submitted through September 30, 1993; and "Nine Mile Point Nuclear Station Safeguards Contingency Plan," with revisions submitted through October 1, 1992. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, Niagara Mohawk Power Corporation shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System, with written followup within 30 days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

G. Niagara Mohawk Power Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment No. 27 and as described in submittals dated March 25, May 7 and 9, June 10 and 25, July 11 and 16, August 19 and 22, September 5, 12, and 23, October 10, 21, and 22, and December 9, 1986, and April 10 and May 20, 1987, and as approved in the SER dated February 1985 (and Supplements 1 through 6) subject to the following provision:

> Niagara Mohawk Power Corporation may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

H. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

I. This license is effective as of the date of issuance and shall expire at midnight on October 31, 2026.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Appendix A - Technical Specifications (NUREG-1253)
2. Appendix B - Environmental Protection Plan

Date of Issuance: July 2, 1987

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 91 TO FACILITY OPERATING LICENSE NPF-69

NINE MILE POINT UNIT 2

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

1.0 INTRODUCTION

Nine Mile Point Unit 2 (NMP2) has been operating with Technical Specifications (TS) issued with the full power operating license (NPF-69) on July 2, 1987, as amended. By application dated October 16, 1998 (NMP2L 1830), as supplemented by letters dated December 30, 1998 (NMP2L 1844), May 10 (NMP2L 1866), June 15 (NMP2L 1872), July 30 (NMP2L 1881), August 2 (NMP2L 1883), August 11 (NMP2L 1885), August 16 (NMP2L 1886), August 19 (NMP2L 1888), August 27 (NMP2L 1893), September 10 (NMP2L 1896), September 30 (NMP2L 1900), December 14, 1999 (NMP2L 1917), January 6 (NMP2L 1923), January 17 (NMP2L 1925), January 21 (NMP2L 1928), January 31 (NMP2L 1929), and February 11, 2000 (NMP2L 1936) Niagara Mohawk Power Corporation (NMPC) the licensee proposed to convert the current Technical Specifications (CTS) to improved Technical Specifications (ITS). The conversion is based upon:

- NUREG-1433, "Standard Technical Specifications for General Electric Plants, BWR/4," Revision 1, dated April 1995,
- NUREG-1434, "Standard Technical Specifications for General Electric Plants, BWR/6," Revision 1, dated April 1995,
- "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," (Final Policy Statement), published on July 22, 1993 (58 FR 39132), and
- 10 CFR 50.36, "Technical Specifications," as amended July 19, 1995 (60 FR 36953).

Hereafter, the proposed or improved TS for NMP2 are referred to as the ITS, the existing TS are referred to as the CTS, and the improved standard TS, such as in NUREG-1433 or NUREG-1434 are referred to as the STS. The corresponding TS Bases are ITS Bases, CTS Bases, and STS Bases, respectively. For convenience, a list of acronyms used in this safety evaluation (SE) is provided in Attachment 1.

In addition to basing the ITS on the STS, the Final Policy Statement, and the requirements in 10 CFR 50.36, the licensee retained portions of the CTS as a basis for the ITS. Plant-specific issues, including design features, requirements, and operating practices, were discussed with the licensee during a series of meetings and telephone conference calls that concluded on December 8, 1999. These plant-specific changes serve to clarify the ITS with respect to the guidance in the Final Policy Statement and STS. Also, based on these discussions, the licensee proposed matters of a generic nature that were not in STS. The NRC staff requested that the licensee submit such generic issues as a proposed change to STS through the NRC/Nuclear Energy Institute's Technical Specifications Task Force (TSTF). These generic issues were considered for specific applications in the NMP2 ITS. Consistent with the Final Policy Statement, the licensee proposed transferring some CTS requirements to licensee-controlled documents (such as the updated final safety analysis report (USAR) for NMP2, for which changes to the documents by the licensee are controlled by a regulation such as 10 CFR 50.59 and may be changed without prior NRC approval). NRC-controlled documents, such as the TS, may not be changed by the licensee without prior NRC approval. In addition, human factors principles were emphasized to add clarity to the CTS requirements being retained in the ITS, and to define more clearly the appropriate scope of the ITS. Further, significant changes were proposed to the CTS Bases to make each ITS requirement clearer and easier to understand.

The overall objective of the proposed amendment, consistent with the Final Policy Statement, is to rewrite, reformat, and streamline the TS for NMP2 to be in accordance with 10 CFR 50.36.

Since the licensee prepared the October 16, 1998, application, a number of amendments to the NMP2 operating license were approved. Table 1 provides the subjects of the amendments and the dates of issuance.

TABLE 1

Amendment No.	Description of Change	Date
84	Update Terminology and References to 50.55a	12/3/98
85	Relocate Monitoring Instruments in TS to USAR	12/22/98
86	Testing of Electrical Protection Assemblies	3/18/99
87	Alternate Isolation Methods for Primary Containment Bypass Leakage Paths or Purge System Pipes	12/16/99
88	New Relief Valves for Drywell Equipment and Drywell Floor Drain Lines	12/16/99
89	Plant Service Water System	2/3/00
90	Reactor Vessel Material Surveillance Program Withdrawal Schedule	2/15/00

The licensee has incorporated these amendments, as appropriate, into the ITS.

The NRC staff's evaluation of the application dated October 16, 1998 (NMP2L 1830), as supplemented by letters dated December 30, 1998 (NMP2L 1844), May 10 (NMP2L 1866), June 15 (NMP2L 1872), July 30 (NMP2L 1881), August 2 (NMP2L 1883), August 11 (NMP2L 1885), August 16 (NMP2L 1886), August 19 (NMP2L 1888), August 27 (NMP2L 1893), September 10

(NMP2L 1896), September 30 (NMP2L 1900), December 14, 1999 (NMP2L 1917),January 6 (NMP2L 1923), January 17 (NMP2L 1925), January 21 (NMP2L 1928), January 31 (NMP2L 1929), and February 11, 2000 (NMP2L 1936), is presented in this SE. The NRC staff issued requests for additional information (RAIs) dated March 26, May 10, May 18, June 16, September 2, and September 21, 1999. Summaries of an NRC staff meeting with the licensee regarding the conversion was issued on December 8, 1999.

The license conditions implementing the conversion will make enforceable the following aspects of the conversion: (1) the relocation of requirements from the CTS and (2) the implementation schedule for new and revised SRs in the ITS.

The Commission's proposed action on the NMP2 application for an amendment dated October 16, 1998, was published in the *Federal Register* on October 20, 1999 (64 FR 56518) and on December 1, 1999 (64 FR 67336). The *Federal Register* notices also addressed beyond-scope issues identified in the licensee's supplemental submittals.

During its review, the NRC staff relied on the Final Policy Statement and the STS as guidance for acceptance of CTS changes. This SE provides a summary basis for the NRC staff's conclusion that the licensee can develop ITS based on STS, as modified by plant-specific changes, and that the use of the ITS is acceptable for continued operation. The SE also explains the NRC staff's conclusion that the ITS, which are based on the STS as modified by plant-specific changes, are consistent with the NMP2 current licensing basis and the requirements of 10 CFR 50.36.

The NRC staff also acknowledges that, as indicated in the Final Policy Statement, the conversion to STS is a voluntary process. Therefore, it is acceptable that the ITS differ from the STS, to reflect the current licensing basis for NMP2. The NRC staff approves the licensee's changes to the CTS with modifications documented in the licensee's supplemental submittals.

For the reasons stated *infra* in this SE, the NRC staff finds that the ITS issued with this license amendment comply with Section 182a of the Atomic Energy Act, 10 CFR 50.36, and the guidance in the Final Policy Statement, and that they are in accord with the common defense and security and provide adequate protection of the health and safety of the public.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act requires that applicants for nuclear power plant operating licenses will state:

> [S]uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TS. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Statement of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports," (33 FR 18610, December 17, 1968). Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

For several years, NRC and industry representatives have sought to develop guidelines for improving the content and quality of nuclear power plant TS. On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, the utility owners groups and the NRC staff developed improved STS, such as NUREG-1433 for GE BWR/4's or NUREG-1434 for GE BWR/6's, that would establish models of the Commission's policy for each primary reactor type. In addition, the NRC staff, licensees, and owners groups developed generic administrative and editorial guidelines in the form of a "Writer's Guide" for preparing TS, which gives greater consideration to human factors principles and was used throughout the development of licensee-specific ITS.

In September 1992, the Commission issued NUREG-1433 and NUREG-1434, which were developed using the guidance and criteria contained in the Commission's Interim Policy Statement. The STS in NUREG-1433 and NUREG-1434 were established as a model for developing the ITS for GE BWR/4 and GE BWR/6 plants in general. The STS reflect the results of a detailed review of the application of the interim policy statement criteria to generic system functions, which were published in a "Split Report" issued to the nuclear steam system supplier owners groups in May 1988. STS also reflect the results of extensive discussions concerning various drafts of STS, so that the application of the TS criteria and the Writer's Guide would consistently reflect detailed system configurations and operating characteristics for all reactor designs. As such, the generic Bases presented in NUREG-1433 and NUREG-1434 provide an abundance of information regarding the extent to which the STS present requirements that are necessary to protect public health and safety. The STS in NUREG-1433 or NUREG-1434 apply to NMP2.

On July 22, 1993, the Commission issued its Final Policy Statement, expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the Act and 10 CFR 50.36 (58 FR 39132). The Final Policy Statement described the safety benefits of the STS, and encouraged licensees to use the STS as the basis for plant-specific TS amendments, and for complete conversions to ITS based on the STS. Further, the Final Policy Statement gave guidance for evaluating the required scope of the TS and defined the guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TS. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General*

Electric Co. (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> [T]here is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

By this approach, existing LCO requirements that fall within or satisfy any of the criteria in the Final Policy Statement should be retained in the TS; those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36 (60 FR 36953, July 19, 1995). The four criteria are as follows:

Criterion 1

Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3

A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4

A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

Part 3.0 of this SE explains the NRC staff's conclusion that the conversion of the NMP2 CTS to ITS based on STS, as modified by plant-specific changes, is consistent with the NMP2 current licensing basis and the requirements and guidance of the Final Policy Statement and 10 CFR 50.36.

3.0 EVALUATION

The NRC staff's ITS review evaluates changes to CTS that fall into nine categories defined by the licensee and includes an evaluation of whether existing regulatory requirements are adequate for controlling future changes to requirements removed from the CTS and placed in licensee-controlled documents.

The NRC staff review also identified the need for clarifications and additions to the October 16, 1998, application in order to establish an appropriate regulatory basis for translation of CTS requirements into ITS. Each change proposed in the amendment request is identified as either a discussion of change (DOC) to the CTS or a justification for difference from the STS. The NRC staff's comments were documented as RAIs and forwarded in letters dated April 9, March 26, May 10, May 18, June 16, September 1, September 2, and September 21, 1999. The licensee provided responses to the RAIs in letters dated December 30, 1998 (NMP2L 1844), May 10 (NMP2L 1866), June 15 (NMP2L 1872), July 30 (NMP2L 1881), August 2 (NMP2L 1883), August 11 (NMP2L 1885), August 16 (NMP2L 1886), August 19 (NMP2L 1888), August 27 (NMP2L 1893), September 10 (NMP2L 1896), September 30 (NMP2L 1900), December 14, 1999 (NMP2L 1917), January 6 (NMP2L 1923), January 17 (NMP2L 1925), January 21 (NMP2L 1928), January 31 (NMP2L 1929), and February 11, 2000 (NMP2L 1936). The letters clarified the licensee's bases for translating the CTS requirements into ITS. The NRC staff finds that the licensee's submittals, including the responses to the RAIs, provide sufficient detail to allow the staff to reach a conclusion regarding the adequacy of the licensee's proposed changes to the CTS.

The license amendment application was organized such that changes were included in each of the following CTS change categories, as appropriate:

1. Administrative Changes, (A) (i.e., nontechnical changes in the presentation of CTS requirements);

2. Technical Changes - More Restrictive, (M) (i.e., new or additional TS requirements);

3. Technical Changes - Less Restrictive (specific), (L) (i.e., changes, deletions, and relaxations of CTS requirements);

4. Technical Changes - Less Restrictive (generic), (LA) (i.e., relocation of details out of the CTS and into the Bases, USAR, QA Manual, or other appropriate licensee-controlled document);

5. Technical Changes - Less Restrictive (generic), (LB) (i.e., the extension of an instrument Completion Time or Surveillance Frequency in accordance with NRC approved vendor topical reports);

6. Technical Changes - Less Restrictive (generic), (LC) (i.e., changes related to the elimination of various instrument requirements, where the instrument is an alarm

or indication-only instrument function that does not otherwise meet the NRC Technical Specification selection criteria);

7. Technical Changes - Less Restrictive (generic), (LD) (i.e., changes that reflect extension of the refueling outage surveillance interval from 18 months to 24 months for surveillances other than channel calibrations);

8. Technical Changes - Less Restrictive (generic), (LE) (i.e., changes that reflect extension of the refueling outage surveillance interval from 18 months to 24 months for channel calibrations surveillances);

9. Relocated Technical Specifications, (R) (i.e., relaxations in which whole CTS specifications (the LCO, and associated actions and SRs) are removed from the CTS (an NRC-controlled document) and placed in licensee-controlled documents).

The changes that are in the ITS conversion for NMP2 for each of the above categories are listed in the following four tables attached to this SE:

- Table A of Administrative Changes to Current Technical Specifications
- Table M of More Restrictive Changes to Current Technical Specifications
- Table L of Less Restrictive Changes to Current Technical Specifications
- Table R of Relocated Specifications and Removed Details from CTS

These tables provide a summary description of the proposed changes to the CTS, the specific CTS that are being changed, and the specific ITS that incorporate the changes. If the table only lists a CTS LCO, as for example LCO 3.4.1, then the CTS being changed is the specific LCO 3.4.1 or the entirety of the specification for LCO 3.4.1 (i.e., LCO, actions, and SRs) is being changed. However, if an action or an SR is listed, then only the specific action or SR is being changed (e.g., LCO 3.4.1, Action a or SR 4.4.1.2). The same is true for an ITS LCO, action or SR, except the ITS is incorporating the change. The tables are only meant to summarize the changes being made to the CTS. The details, as to what the actual changes are and how they are being made to the CTS or ITS, are only provided in the licensee's application and supplemental letters.

The general categories of changes to the licensee's CTS requirements and STS differences may be better understood as follows:

A. Administrative Changes

Administrative (nontechnical) changes are intended to incorporate human factors principles into the form and structure of the ITS so that plant operations personnel can use them more easily. These changes are editorial in nature or involve the reorganization or reformatting of CTS requirements without affecting technical content or operational restrictions. Every section of the ITS reflects this type of change. In order to ensure consistency, the NRC staff and the licensee have used the STS as guidance to reformat and make other administrative changes. Among the changes proposed by the licensee and found acceptable by the NRC staff are:

1. Identifying plant-specific wording for system names, etc.;

2. Splitting up requirements currently grouped under a single current specification to more appropriate locations in two or more specifications of ITS;

3. Combining related requirements currently presented in separate specifications of the CTS into a single specification of ITS;

4. Presentation changes that involve rewording or reformatting for clarity (including moving an existing requirement to another location within the TS) but which do not involve a change in requirements;

5. Wording changes and additions that are consistent with CTS interpretation and practice, and that more clearly or explicitly state existing requirements;

6. Deletion of TS whose applicability has expired; and

7. Deletion of redundant TS requirements that exist elsewhere in the TS.

Table A lists the administrative changes being made in the NMP2 ITS conversion. Table A is organized in ITS order by each A-type DOC to the CTS, and provides a summary description of the administrative change that was made, and CTS and ITS references. The NRC staff reviewed all of the administrative and editorial changes proposed by the licensee and finds them acceptable because they are compatible with the Writer's Guide and STS, do not result in any change in operating requirements, and are consistent with the Commission's regulations.

B. Technical Changes - More Restrictive

The licensee, in electing to implement the specifications of the STS, proposed a number of requirements more restrictive than those in the CTS. The ITS requirements in this category include requirements that are either new, more conservative than corresponding requirements in the CTS, or that have additional restrictions that are not in the CTS but are in the STS. Examples of more restrictive requirements are placing an LCO on plant equipment which is not required by the CTS to be operable, more restrictive requirements to restore inoperable equipment, and more restrictive SRs. Table M lists the more restrictive changes being made in the NMP2 ITS conversion. Table M is organized in ITS order by each M-type DOC to the CTS and provides a summary description of the more restrictive change that was adopted, and the CTS and ITS references. These changes are additional restrictions on plant operation that enhance safety and are acceptable.

C. Technical Changes - Less Restrictive

Less restrictive requirements include deletions and relaxations to portions of the CTS requirements that are being retained in ITS. When requirements have been shown to give little or no safety benefit, their relaxation or removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of: (1) generic NRC actions, (2) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the owners groups comments on the STS. The NRC staff reviewed generic relaxations contained in the STS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The NMP2 design was also reviewed to determine if the specific design basis and licensing basis for NMP2 are consistent with the technical basis for the model requirements in the STS, and thus provide a basis for the ITS.

All of the less restrictive changes to the CTS have been evaluated and found to involve deletions and relaxations to portions of the CTS requirements that can grouped in the following ten categories:

(1) Relaxation of LCOs and Administrative Controls (Category I)

(2) Relaxation of Applicability (Category 2)

(3) Relaxation of Surveillance Requirement (Category 3)

(4) Relaxation of Required Action Detail (Category 4)

(5) Relaxation of Required Actions to exit Applicability (Category 5)

(6) Relaxation of Completion Time (Category 6)

(7) Allow Mode changes when LCO not met (Category 7)

(8) Elimination of the requirement to lock the Reactor Mode switch in Shutdown or Refuel (Category 8)

(9) Elimination of CTS Reporting Requirements (Category 9)

(10) Relaxation of Fuel Cycle from 18 to 24 Months (Category 10)

The following discussions address why portions of various specifications within each of these ten categories of information or specific requirements are not required to be included in ITS.

(1) Relaxation of LCOs and Administrative Controls (Category 1)

CTS contain LCOs that are overly restrictive because they specify limits on operational and system parameters and on system Operability beyond those necessary to meet safety analysis assumptions. CTS also contain administrative controls that do not contribute to the safe operation of the plant. The ITS, consistent with the guidance in the STS, omit such operational limits and administrative controls. This category of change includes (1) deletion of equipment or systems addressed by the CTS LCOs which are not required or assumed to function by the applicable safety analyses; (2) addition of explicit exceptions to the CTS LCO requirements, consistent with the guidance of the

STS and normal plant operations, to provide necessary operational flexibility but without a significant safety impact; and (3) deletion of miscellaneous administrative controls such as reporting requirements—sometimes contained in action requirements—that have no affect on safety. Deletion of such administrative controls allows operators to more clearly focus on issues important to safety. The ITS LCOs and administrative controls resulting from these changes will continue to maintain an adequate degree of protection consistent with the safety analysis while providing an improved focus on issues important to safety and necessary operational flexibility without adversely affecting the safe operation of the plant. Therefore, less restrictive changes falling within Category 1 are acceptable.

(2) Relaxation of Applicability (Category 2)

Reactor operating conditions are used in CTS to define when the LCO features are required to be operable. CTS applicabilities can be specific defined terms of reactor conditions: hot shutdown, cold shutdown, reactor critical or power operating condition. Applicabilities can also be more general. Depending on the circumstances, CTS may require that the LCO be maintained within limits in "all modes" or "any operating mode." Generalized applicability conditions are not contained in STS, therefore ITS eliminate CTS requirements such as "all modes" or "any operating mode," replacing them with ITS defined modes or applicable conditions that are consistent with the application of the plant safety analysis assumptions for operability of the required features.

In another application of this type of change, CTS requirements may be eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements that are indeterminant or which are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the TS are satisfied by exiting the applicability thus taking the plant out of the conditions that require the safety system to be operable. These changes are consistent with STS and changes specified as Category 2 are acceptable.

(3) Relaxation of Surveillance Requirement (Category 3)

Prior to placing the plant in a specified operational mode or other condition stated in the Applicability of an LCO, and in accordance with the specified SR Frequency thereafter, the CTS require verifying the Operability of each LCO-required component by meeting the SRs associated with the LCO. This usually entails performance of testing to demonstrate the Operability of the LCO-required components, or the verification that specified parameters are within LCO limits. A successful demonstration of Operability requires meeting the specified acceptance criteria as well as any specified conditions for the conduct of the test. Relaxations of CTS SRs include relaxing both the acceptance criteria and the conditions of performance. These CTS SR relaxations are consistent with STS.

Relaxations of CTS SR acceptance criteria provide operational flexibility, consistent with the guidance of the STS, but do not reduce the level of assurance of Operability provided by the successful performance of the surveillance. Such revised acceptance criteria are acceptable because they remain consistent with the application of the plant safety analysis assumptions for Operability of the LCO-required features.

Relaxations of CTS SR performance conditions include not requiring testing of deenergized equipment (e.g., instrumentation Channel Checks) and equipment that is already performing its intended safety function (e.g., position verification of valves locked in their safety actuation position). These changes are acceptable because the existing surveillances are not necessary to ensure the capability of the affected components to perform their intended functions. Another relaxation of SR performance conditions is the allowance to verify the position of valves in high radiation areas by administrative means. This change is acceptable because the TS administrative controls (ITS 5.7) regarding access to high radiation areas make the likelihood of mispositioning such valves negligible.

Finally, the ITS permits the use of an actual as well as a simulated actuation signal to satisfy SRs for automatically actuated systems. This is acceptable because TS required features cannot distinguish between an "actual" signal and a "test" signal.

These relaxations of CTS SRs optimize test requirements for the affected safety systems and increase operational flexibility. Therefore, because of the reasons stated, less restrictive changes falling within Category 3 are acceptable.

(4) Relaxation of Required Action Detail (Category 4)

CTS provides lists of acceptable devices that may be used to satisfy LCO requirements. The ITS reflect the STS approach to provide LCO requirements that specify the protective limit that is required to meet safety analysis assumptions for required features. The protective limits replace the lists of specific devices previously found to be acceptable to the NRC staff for meeting the LCO. The ITS changes provide the same degree of protection required by the safety analysis and provide flexibility for meeting limits without adversely affecting operations since equivalent features are required to be operable. These changes are consistent with STS and changes specified as Category 4 are acceptable.

(5) Relaxation of Required Actions to exit Applicability (Category 5)

CTS require that in the event specified LCOs are not met, penalty factors to reactor operation, such as resetting setpoints, and power reductions shall be initiated as the method to reestablish the appropriate limits. The ITS are constructed to specify actions for conditions of required features made inoperable. Adopting ITS action requirements for exiting LCO applicabilities is acceptable because the plant remains within analyzed parameters by performance of required actions, or the actions are constructed to minimize risks associated with continued operation while providing time to repair inoperable features. Such actions add margin to safety or verify equipment status such as interlock status for the mode of operation, thereby providing assurance that the plant is configured appropriately or operations that could result in a challenge to safety systems are exited in a time period that is commensurate with the safety importance of the system. Additionally, other changes to TS actions include placing the reactor in a mode where the specification no longer applies, usually resulting in an extension to the time period for taking the plant into shutdown conditions. These actions are commensurate with industry standards for reductions in thermal power in an orderly fashion without compromising safe operation of the plant. These changes are consistent with STS and changes specified as Category 5 are acceptable.

(6) Relaxation of Completion Time (Category 6)

Upon discovery of a failure to meet an LCO, STS specify times for completing required Actions of the associated TS conditions. Required Actions of the associated conditions are used to establish remedial measures that must be taken within specified completion times (allowed outage times). These times define limits during which operation in a degraded condition is permitted.

Adopting completion times from the STS is acceptable because completion times take into account the operability status of the redundant systems of TS required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a design basis accident (DBA) occurring during the repair period. These changes are consistent with STS, and allowed outage time extensions specified as Category 6 are acceptable.

(7) Allow Mode changes when LCO not met (Category 7)

Making LCO 3.0.4 not applicable (allowing Mode changes when LCO is not met for specific conditions) is acceptable because the requirements which are not met either perform a function which does not result in the inoperability of any functions which are directly credited in the mitigation of an event or other redundant components are available such that a single failure would not result in a loss of function. Therefore, in these instances it is acceptable to eliminate the requirements of LCO 3.0.4.

(8) Elimination of the requirement to lock the Reactor Mode switch in Shutdown or Refuel (Category 8)

Elimination of the requirement to "lock" the Reactor Mode switch in "Shutdown" or "Refuel" is acceptable because the Required Action still requires that the reactor mode switch be maintained in the shutdown or refuel position when the specified condition is not met and Technical Specification requirements will prevent withdrawal of control rods until the LCO requirements are restored. Therefore, it is acceptable to eliminate the requirement to "lock" the Mode switch in shutdown or Refuel.

(9) Elimination of CTS Reporting Requirements (Category 9)

CTS include requirements to submit Special Reports when specified limits are not met. Typically, the time period for the report to be issued is within 30 days. However, the STS eliminates the TS administrative control requirements for Special Reports and instead relies on the reporting requirements of 10 CFR 50.73. ITS changes to reporting requirements are acceptable because 10 CFR 50.73 provides adequate reporting requirements, and the special reports do not affect continued plant operation. Therefore, this change has no impact on the safe operation of the plant. Additionally, deletion of TS reporting requirements reduces the administrative burden on the plant and allows efforts to be concentrated on restoring TS required limits. These changes are consistent with STS and changes specified as Category 9 are acceptable.

(10) Relaxation of Fuel Cycle from 18 to 24 Months (Category 10)

The ITS includes changes to the frequency of the current surveillance requirements (SRs) to accommodate the planned change from an 18 to 24-month fuel cycle. In its submittal, the licensee stated that the proposed modifications are based on the guidance provided by the staff in Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. In the GL 91-04 letter, it is stated that the NRC staff has reviewed a number of requests to extend 18-month surveillances to the end of a fuel cycle and a few requests for changes in surveillance interval to accommodate a 24-month fuel cycle. The staff has found that the effect on safety is small because safety systems use redundant electrical and mechanical components and because licensees perform other surveillances during plant operation that confirm that these systems and components can perform their safety functions. In applying GL 91-04, the licensee evaluated the effect on safety of an increase in 18-month surveillance intervals to accommodate a 24-month fuel cycle. In its submittal, the licensee stated that data was collected to perform an analysis using the Microsoft Excel 7.0, SYSAT 7.01 and Instrument History Performance Analysis programs by CRS Engineering, Inc. The results of the computer analysis and a review of past test and surveillance data indicated that the system performance during the surveillance test was found to be within acceptable limits. No system failures were identified that would have adversely impacted safety during the period reviewed. The licensee further stated that, based on the results of their evaluation, system redundancy, detectability of the failures by other mid-cycle testing and equipment performance during these mid-cycle tests, it was concluded that the proposed test interval extensions have an insignificant effect on safety. Also, the licensee confirmed that assumptions in the plant licensing basis would not be invalidated on the basis of performing any surveillance at the bounding surveillance interval limit provided to accommodate a 24-month fuel cycle.

Based on the above, the staff concludes that, in accordance with GL 91-04, the proposed changes have a negligible effect on safety. Historical data supports this conclusion. The proposed changes follow the guidance of GL 91-04, and there are no plant-specific circumstances that preclude 24-month testing intervals. Therefore, based on conformance with the GL 91-04 criteria, the proposed changes specified as Category 10 are acceptable.

D. Relocated Less Restrictive Requirements (Not Entire Specifications)

When requirements have been shown to give little or no safety benefit, their removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups comments on STS. The NRC staff reviewed generic relaxations contained in STS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The design was also reviewed to determine if the specific design basis and licensing basis are consistent with the technical basis for the model requirements in STS, and thus provide a basis for ITS. Changes to the CTS that involve the removal of specifications, specific requirements and detailed information from individual specifications were all evaluated and grouped within the following Types 1 through 5:

Type 1	Details of System Design and System Description Including Design Limits
Type 2	Descriptions of Systems Operation
Type 3	Procedural Details for TS Requirements and Related Reporting Requirements
Type 4	Performance Requirements for Indication-only Instrumentation and Alarms
Type 5	Relocated Specifications/Surveillance Requirement/Administrative Controls Requirement

The following discussions address why each of the five types of information or specific requirements are not required to be included in ITS .

Details of System Design and System Description Including Design Limits *(Type 1)*

The design of the facility is required to be described in the USAR by 10 CFR 50.34. In addition, the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50 require that plant design be documented in controlled procedures and drawings, and maintained in accordance with an NRC-approved QA Program (USAR Chapter 17). In 10 CFR 50.59, controls are specified for changing the facility as described in the USAR which includes the new Technical Requirements Manual (TRM) by reference, and in 10 CFR 50.54(a) criteria are specified for changing the QA Program. In the ITS, the Bases also contain descriptions of system design. The NMP2 administrative controls specification ITS 5.5.10 specifies controls for changing the Bases. Removing details of system design from the CTS is acceptable because this information will be adequately controlled by NRC requirements, the USAR, controlled design documents and drawings, or the TS Bases, as appropriate. Cycle-specific design limits are moved from the CTS to the Core Operating Limits Report (COLR) in accordance with Generic Letter 88-16. ITS Administrative Controls are revised to include the programmatic requirements for controlling the COLR.

Descriptions of Systems Operation *(Type 2)*

The plans for the normal and emergency operation of the facility are required to be described in the USAR by 10 CFR 50.34. ITS 5.4.1.a requires written procedures to be established, implemented, and maintained for plant operating procedures including procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Controls specified in 10 CFR 50.59 apply to changes in procedures as described in the USAR. In the ITS, the Bases also contain descriptions of system operation. It is acceptable to remove details of system operation from the TS because this type of information will be adequately controlled in the USAR, plant operating procedures, and the TS Bases, as appropriate.

Procedural Details for Meeting TS Requirements & Related Reporting Requirements(Type 3)

Details for performing action and surveillance requirements are more appropriately specified in the plant procedures required by ITS 5.4.1, the USAR, and the ITS Bases. For example, control of the plant conditions appropriate to perform a surveillance test is an issue for procedures and scheduling and has previously been determined to be unnecessary as a TS restriction. As indicated in Generic Letter 91-04, allowing this procedural control is consistent with the vast majority of other SRs that do not dictate plant conditions for surveillances. Prescriptive procedural information in an action requirement is unlikely to contain all procedural considerations necessary for the plant operators to complete the actions required, and referral to plant procedures is therefore required in any event. Other changes to procedural details include those associated with limits retained in the ITS. For example, the ITS requirement may refer to programmatic requirements such as COLR, included in ITS Section 5.5, which specifies the scope of the limits contained in the COLR and mandates NRC approval of the analytical methodology. The QA Program is approved by the NRC and contained in USAR Chapter 17, and changes to the QA Program are controlled by 10 CFR 50.54(a). The Offsite Dose Calculation Manual (ODCM) is required by ITS section 5.5.1. The TRM is incorporated by reference into the USAR, and changes to the TRM are controlled by 10 CFR 50.59. The Inservice Test (IST) program is required by ITS 5.5.6 and is controlled by ITS 5.4.1.e.

The removal of these kinds of procedural details from the CTS is acceptable because they will be adequately controlled by NRC requirements, the USAR, plant procedures, Bases and COLR, as appropriate. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Similarly, removal of reporting requirements from LCOs is appropriate because ITS 5.6, 10 CFR 50.36 and 10 CFR 50.73 adequately cover the reports deemed to be necessary.

Performance Requirements for Indication-Only Instrumentation and Alarms *(Type 4)*

Indication-only instrumentation, test equipment, and alarms are usually not required to be operable to support TS operability of a system or component unless these items are included in TS as Accident Monitoring instrumentation. Thus, with the exception of the Accident Monitoring instrumentation, STS do not include operability requirements for indication-only equipment. The availability of such indication instruments, monitoring instruments, and alarms, and necessary compensatory activities if they are not available, are more appropriately specified in plant operational, maintenance, and annunciator response procedures required by ITS 5.4.1. Removal of requirements for indication-only instrumentation and alarms from the CTS is acceptable because they will be adequately controlled in plant procedures.

Relocated Specifications/Surveillance Requirement/Administrative Controls Requirement *(Type 5)*

Table R lists the requirements and detailed information in the CTS that are being moved to licensee-controlled documents and not retained in the ITS. Table R is organized in ITS order by each LA, LC and R-type DOC to the CTS (the R DOCs are described in Section 3.E below). It includes the following: (1) the ITS section or specification designation, as appropriate, followed by the DOC identifier (e.g., 3.1.1 followed by LA.1 means ITS Specification 3.1.1, DOC LA.1); (2) CTS reference; (3) a summary description of the relocated details; (4) the name of the document to contain the relocated details or requirements (new location); (5) the regulation (or ITS section) for controlling future changes to relocated requirements (control process); and (6) a characterization of the type of change.

The NRC staff has concluded that these types of detailed information and specific requirements do not need to be included in the ITS to ensure the effectiveness of ITS to adequately protect the health and safety of the public. Accordingly, these requirements may be moved to one of the following licensee-controlled documents for which changes are adequately governed by a regulatory or TS requirement:

- TS Bases controlled in accordance with ITS 5.5.10, "Technical Specifications (TS) Bases Control Program."
- USAR (which includes the TRM) controlled by 10 CFR 50.59.
- Programmatic documents required by ITS Section 5.5 controlled by ITS Section 5.4
- Inservice Inspection (ISI) and Inservice Testing Programs controlled by 10 CFR 50.55a
- Offsite Dose Calculation Manual controlled by ITS 5.5.1.
- Core Operating Limits Report controlled by ITS 5.6.5.
- QA plan, as approved by the NRC and referenced in the USAR, controlled by 10 CFR Part 50, Appendix B, and 10 CFR 50.54(a).
- Site Emergency Plan controlled by 10 CFR 50.54(q).

To the extent that information has been relocated to licensee-controlled documents, such information is not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, where such information is contained in LCOs and associated requirements in the CTS, the NRC staff has concluded that they do not fall within any of the four criteria contained in 10 CFR 50.36 and discussed in the Final Policy Statement (see Section 2.0 of this SE). Accordingly, existing detailed information, such as generally described above, may be removed from the CTS and not included in the ITS.

E. Relocated Entire CTS Specifications

The Final Policy Statement states that LCOs and associated requirements that do not satisfy or fall within any of the four specified criteria (now contained in 10 CFR 50.36) may be relocated from existing TS (an NRC-controlled document) to appropriate licensee-controlled documents. This section of the SE discusses the relocation of entire specifications in the CTS to licensee-controlled documents. These specifications include the LCOs, Action Statements (i.e., Actions), and associated SRs. In its application and its supplements, the licensee proposed relocating such specifications from the CTS to the USAR, which includes the TRM, the Process Control Program (PCP), and the ODCM, as appropriate. The staff has reviewed the licensee's submittals, and finds that relocation of these requirements to the USAR, TRM, PCP, and ODCM is acceptable in that changes to the USAR, TRM, PCP, and ODCM will be adequately controlled by 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55a, and ITS 5.5.1 as applicable. These provisions will continue to be implemented by appropriate station procedures (i.e., operating procedures, maintenance procedures, surveillance and testing procedures, and work control procedures).

Table R lists all specifications that are being relocated from the CTS to licensee-controlled documents. Table R is organized by each R, LA and LC-type DOC to the CTS, in a manner generally consistent with the organization of requirements in the ITS (the LA and LC DOCs are described in Section 3.D above). Table R includes: (1) references to the DOC, (2) references to the relocated CTS requirements, (3) summary descriptions of the relocated CTS requirements, (4) names of the documents that will contain the relocated requirements (i.e., the new location), (5) the methods for controlling future changes to the relocated requirements (i.e., the control process), and (6) a characterization of the type of item relocated.

The NRC staff's evaluation of each relocated specification listed in Table R is provided below, mostly in CTS order.

1. 3/4.3.2.2.h RCIC Drywell Pressure - High

The function of the RCIC Drywell Pressure - High Function is to provide an isolation signal to the RCIC turbine exhaust inboard and outboard vacuum breaker isolation valves. A high drywell pressure signal in conjunction with a RCIC low steam line pressure signal will isolate the valves. The isolation of these portions of the RCIC system is not used to mitigate a design basis accident or transient. The isolation is provided for protection of the RCIC turbine exhaust lines against operation at high pressures which might cause damage to the equipment. Credit for this isolation is not assumed in any design basis analyses. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the RCIC Drywell Pressure - High Function LCO and Surveillances may be relocated to the TRM.

2. CTS 3/4.3.3.A.2.f ADS 'A' - Manual Inhibit and CTS 3/4.3.3.B.2.e ADS 'B' - Manual Inhibit.

The ADS Manual Inhibit switch allows the operator to defeat ADS actuation as directed by the emergency operating procedures under conditions for which ADS would not be desirable. For example, during an ATWS event low pressure ECCS system activation would dilute sodium pentaborate injected by the Standby Liquid Control (SLC) System thereby reducing the effectiveness of the SLC System ability to shutdown the reactor. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the portions of the LCO and Surveillances applicable to the ADS Manual Inhibit switch may be relocated to the TRM.

3. CTS 3/4.3.6.2 Source Range Monitor

The Source Range Monitor (SRM) control rod block functions to prevent a control rod withdrawal error during reactor startup utilizing SRM signals to create the rod block signal. SRM signals are used to monitor neutron flux during refueling, shutdown, and startup conditions. No design basis accident (DBA) or transient analysis takes credit for rod block signals initiated by the SRMs. The SRM control rod block function does not meet any of the criteria in 10 CFR 50.36 and may be relocated out of the CTS to the TRM.

4. CTS 3/4.3.6.3 Intermediate Range Monitor

The Intermediate Range Monitor (IRM) control rod block functions to prevent a control rod withdrawal error during reactor startup utilizing IRM signals to create the rod block signal. IRMs are provided to monitor the neutron flux levels during refueling, shutdown, and startup conditions. No design basis accident (DBA) or transient analysis takes credit for rod block signals initiated by IRMs. The IRM control rod block function does not meet any of the criteria in 10 CFR 50.36 and may be relocated out of the CTS to the TRM.

5. CTS 3/4.3.6.4 Scram Discharge Volume

The Scram Discharge Volume (SDV) control rod block functions to prevent control rod withdrawals, utilizing SDV signals to create the rod block signal if water is accumulating in the SDV. The purpose of measuring the SDV water level is to ensure that there is sufficient volume remaining to contain the water discharged by the control rod drives during a scram, thus ensuring that the control rods will be able to insert fully. This rod block signal provides an indication to the operator that water is accumulating in the SDV and prevents further rod withdrawals. With continued water accumulation, a reactor protection system initiated scram signal will occur. Thus, the SDV water level rod block signal provides an opportunity for the operator to take action to avoid a subsequent scram. No design basis accident (DBA) or transient takes credit for rod block signals initiated by the SDV instrumentation. The SDV rod block signals do not meet any of the criteria in 10 CFR 50.36 and may be relocated out of the CTS to the TRM.

6. CTS 3/4.3.6.5 Reactor Coolant System Recirculation Flow

An increase in reactor recirculation flow causes an increase in neutron flux which results in an increase in reactor power. However, this increase in neutron flux is monitored by the neutron monitoring system which has the capability of providing a reactor scram, when required. No design basis accident (DBA) or transient analysis takes credit for rod block signals initiated by the reactor coolant recirculation system. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Control Rod Block LCO and Surveillances applicable to RCS recirculation flow instrumentation may be relocated to the TRM.

7. CTS 3/4.3.7.1.2 Area Monitors - Criticality Monitor (New Fuel Storage Vault) and Control Room Direct Radiation Monitor

The area radiation monitors are used to indicate when the radiation in the area has exceeded its allowable setpoint. There are no automatic functions that are performed by these instruments. The instruments are not used to mitigate a design basis accident (DBA) or transient. Information provided by these instruments on the radiation levels would have limited or no use in identifying/assessing core damage. The area radiation monitors do not meet any of the criteria in 10 CFR 50.36 and may be relocated out of the CTS to the TRM.

8. CTS 3/4.3.7.2 Seismic Monitoring Instrumentation

In the event of an earthquake, seismic monitoring instrumentation is required to determine the magnitude of the seismic event. These instruments do not perform any automatic action. They are used to measure the magnitude of the seismic event for comparison to the design basis of the plant to ensure the design margins for plant equipment and structures have not been violated. Since the determination of the magnitude of the seismic event is performed after the event has occurred, this instrumentation has no bearing on the mitigation of any design basis accident (DBA) or transient. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Seismic Monitoring Instrumentation LCO and Surveillances may be relocated to the TRM.

9. CTS 3/4.3.7.5 Accident Monitoring Instrumentation

Each individual accident monitoring parameter has a specific purpose, however, the general purpose for all accident monitoring instrumentation is to provide sufficient information to confirm an accident is proceeding per prediction, i.e. automatic safety systems are performing properly, and deviations from expected accident course are minimal.

The NRC position on application of the deterministic criteria to post-accident monitoring instrumentation is documented in letter dated May 7, 1988 from T.E. Murley (NRC) to R.F. Janecek (BWROG). The position was that the post-accident monitoring instrumentation table list should contain, on a plant specific basis, all Regulatory Guide 1.97 Type A instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the NMP2 Regulatory Guide 1.97 instruments. Those instruments meeting these criteria have remained in Technical Specifications. The instruments not meeting these criteria and also not meeting the criteria of 10 CFR 50.36 have been relocated from the Technical Specifications to plant controlled documents. Therefore, the suppression chamber air temperature instruments, which are neither Type A nor a Category 1 instrument have been relocated to the TRM.

10. CTS 3/4.3.7.7 Traversing In-core Probe System

The traversing in-core probe (TIP) system is used for calibration of the LPRM detectors. The TIP system is positioned axially and radially throughout the core to calibrate the local power range monitors (LPRMs). When not in use the TIP instruments are retracted into a storage position outside the drywell. The TIP system supports the operability of the LPRMs. With LPRM operability addressed there is no need to address the TIP system in the Technical

Specifications. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the TIP System LCO and Surveillances may be relocated to the TRM.

11. CTS 3/4.3.7.8 Loose-part Detection System

The loose-part detection system is used to detect loose parts in the reactor vessel. The instrumentation does not indicate that there is a degradation in the primary pressure boundary but indicates that there might be a remote chance of damage to a component due to a loose part. Fuel failure due to fuel bundle flow blockage from a lost part will be detected by the radiation monitors in the offgas stream. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Loose-Part Detection System LCO and Surveillances may be relocated to the TRM.

12. CTS 3/4.3.7.9 Radioactive Liquid Effluent Monitoring Instrumentation

The radioactive liquid effluent monitoring instrumentation is neither a safety system nor is connected to the reactor coolant system. This instrumentation is used for the purpose of showing conformance to the discharge limits of 10 CFR part 20. It is not installed to detect excessive reactor coolant leakage. The radioactive liquid effluent monitors are used routinely to provide continuous check on the release of radioactive liquid effluent from the normal plant liquid effluent flowpaths. These Technical Specifications require the Licensee to maintain operability of various liquid effluent monitors and establish setpoints in accordance with the Offsite dose Calculation Manual (ODCM). The Alarm/Trip setpoints are established to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. Plant DBA analyses do not assume any action, either automatic or manual, resulting from radioactive effluent monitors. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Radioactive Liquid Effluent Monitoring Instrumentation LCO and Surveillances may be relocated to the ODCM.

13. CTS 3/4.3.7.10 Radioactive Gaseous Effluent Monitoring Instrumentation

The radioactive gaseous effluent monitoring instrumentation is neither a safety system nor is it connected to the reactor coolant system. The primary function of this instrumentation is to show conformance to the discharge limits of 10 CFR Part 20. This instrumentation is not installed to detect excessive reactor coolant leakage. The radioactive gaseous effluent monitors are used routinely to provide continuous check on the releases of radioactive gaseous effluents from the normal plant gaseous effluent flowpaths. These Technical Specifications require the Licensee to maintain operability of various effluent monitors and establish setpoints in accordance with the ODCM. The alarm/trip setpoint are established to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. Plant DBA analyses do not assume any action, either automatic or manual, resulting from radioactive effluent monitors. In addition, the explosive gas monitor instrumentation is provided to ensure that the concentration of potentially explosive gas mixtures contained in the gaseous radwaste treatment system is adequately monitored, which will help ensure that the concentration is maintained below the flammability limit of hydrogen. However, any detonation in the offgas system will not affect the function of any safety related equipment. The concentration of hydrogen in the offgas stream is not an initial assumption of any design basis accident (DBA) or transient analysis. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Radioactive Gaseous Effluent Monitoring Instrumentation LCO and Surveillances may be relocated to the ODCM.

14. CTS 3/4.3.9.2 Service Water System

The function of the Service Water System instrumentation channels is to either ensure the Ultimate Heat Sink is functioning following an earthquake or other non-design basis event, to ensure that indication is available to perform surveillances, or to ensure the intake structure deicer heater system operates automatically. No design basis analysis takes credit for any of these instruments. In addition, other Technical Specifications continue to ensure that the intake deicer heaters are Operable when required, and an SR will continue to ensure that the service water supply header discharge temperature is within limits (thus an indicator must be Operable to measure the temperature). None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Service Water System instrumentation LCO and Surveillances may be relocated to the TRM.

15. CTS 3/4.4.4 Chemistry

Poor reactor coolant water chemistry may contribute to the long term degradation of system materials and thus is not of immediate importance to the plant operator. Reactor coolant water chemistry is monitored for a variety of reasons. One reason is to reduce the possibility of failures in the reactor coolant system pressure boundary caused by corrosion. Severe chemistry transients have resulted in failure of thin walled LPRM instrument dry tubes in a relatively short period of time. However, these LPRM dry tube failures result in loss of the LPRM function and are readily detectable. In summary, the chemistry monitoring activity serves a long term preventative rather than mitigative purpose. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Reactor Coolant System Chemistry LCO and Surveillances may be relocated to the TRM.

16. CTS 3/4.4.8 Structural Integrity

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the components life. Other Technical Specifications require important systems to be operable (for example, ECCS 3/4.5.1) and in a ready state for mitigative action. This Technical Specification is more directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. Hence it is not necessary to retain this specification to ensure immediate operability of safety systems. Further, this Technical Specification prescribes inspection requirements which are performed during plant shutdown. It therefore does not directly address the response to design basis accidents (DBA). None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Structural Integrity LCO and Surveillances may be relocated to the TRM.

17. CTS 3/4.7.2 Revetment-Ditch Structure

The purpose of the Revetment-Ditch Structure is to protect the plant fill and foundation from wave erosion, expected during the probable maximum windstorm for a maximum still water elevation of 254 feet. A windstorm is not a design basis accident or transient, thus the Revetment-Ditch Structure is not credited in any safety analysis. In addition, the Revetment-Ditch Structure can sustain a high degree of damage and still perform its function. The

Revetment-Ditch Structure Technical Specification requirements were put in place to ensure that severe damage will not go undetected for a substantial period of time and if severe damage occurs, facility actions will be taken to repair the Revetment-Ditch Structure. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Revetment-Ditch Structure LCO and Surveillances may be relocated to the TRM.

18. CTS 3/4.7.6 Sealed Source Contamination

The limitations on sealed source contamination are intended to ensure that the total body or individual organ irradiation doses does not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of $\leq$ 0.005 microcuries of removable contamination on each sealed source. This requirement and the associated Surveillance Requirements bear no relation to the conditions or limitations which are necessary to ensure safe reactor operation. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Sealed Source Contamination LCO and Surveillances may be relocated to the TRM.

19. CTS 3/4.8.4.1 AC Circuits Inside Primary Containment

The circuits involved in this LCO are kept normally de-energized and do not participate in plant safety actions. These circuits are primarily for lighting, utility outlets and convenient power plugs, to be used in the event of plant walkdowns, maintenance and in-situ test and/or observations. Therefore, they are of a non-Class 1E nature. They are properly separated from all other Class 1E circuits and operation or failure of these non-Class 1E circuits do not impose any degradation on Class 1E circuits. Thus, in any event, these circuits have no impact on plant safety systems. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the AC Circuits Inside Primary Containment LCO and Surveillances may be relocated to the TRM.

20. CTS 3/4.8.4.2 Primary Containment Penetration Conductor Overcurrent Protective Devices and CTS 3/4.8.4.3 Emergency Lighting System - Overcurrent Protective Devices

The primary feature of these protective devices is to open the control and/or power circuit whenever the load conditions exceed the present current demands. This is to protect the circuit conductors against damage or failure due to overcurrent heating effects. The continuous monitoring of the operating status of the overcurrent protection devices is impracticable and not covered as part of the control room monitoring, except after trip condition indication. In the event of failure of this protective device to trip the circuit, the upstream protective device is expected to operate and isolate the faulty circuit. Thus, the upper level (back-up) protection will prevent loss of redundant power source. In the worst case fault condition, a single division of protective functions can be lost. However, this scenario is covered under a single failure criterion. The overcurrent protection devices ensure the pressure integrity of the containment penetration. With failure of the device it is postulated that the wire insulation will degrade resulting in a containment leak path during a LOCA. However, containment leakage is not a process variable and is not considered as part of the primary success path. Containment penetration degradation will be identified during the normal containment leak rate tests required by 10 CFR Part 50, Appendix J. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Primary Containment Penetration Conductor and Emergency Lighting System Overcurrent Protective Devices LCOs and Surveillances may be relocated to the TRM.

21. CTS 3/4.9.1.b.4 Reactor Mode Switch — Fuel Grapple Position and CTS 3/4.9.6 Refueling Platform

Operability of the refueling platform equipment (crane, main hoist including fuel grapple position, and auxiliary hoist) ensures that only the hoists of the refueling platform will be used to handle fuel within the reactor pressure vessel, hoists have sufficient load capacity for handling fuel assemblies and/or control rods and the core internals and pressure vessel are protected from excessive lifting force if they are inadvertently engaged during lifting operations. Although the interlocks designed to provide the above capabilities can prevent damage to the refueling platform equipment and core internals, they are not assumed to function to mitigate the consequences of a design basis accident. Further, in analyzing the control rod withdrawal error during refueling, if any one of the operations involved in initial failure or error is followed by any other single equipment failure or single operator error, the necessary safety actions are taken (e.g., rod block or scram) automatically prior to violation of any limits. Hence the refueling platform interlocks are not part of the primary success path in mitigating the control rod withdrawal error during refueling. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Reactor Mode Switch— Fuel Grapple Position and Refueling Platform LCOs and Surveillances may be relocated to the TRM.

22. CTS 3/4.9.5 Communications

Communication between the control room and refueling floor personnel is maintained to ensure that refueling personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling. The communications allow for coordination of activities that require interaction between the control room and refueling floor personnel (such as the insertion of a control rod prior to loading fuel). However, the refueling system design accident or transient response does not take credit for communications. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Communications LCOs and Surveillances may be relocated to the TRM.

23. CTS 3/4.9.7 Crane Travel - Spent Fuel Storage Pool

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event the load is dropped, the activity release will be limited to that contained in a single fuel assembly and any possible distortion of the fuel in the storage racks will not result in a critical array. Administrative monitoring of loads moving over the fuel storage racks serves as a backup to the crane interlocks. Although this Technical Specification supports the maximum refueling accident assumption in the design basis accident (DBA), the crane travel limits are not monitored and controlled during operation; they are checked on a periodic basis to ensure operability. The deterministic criteria for Technical Specification retention are, therefore, not satisfied. None of the 10 CFR 50.36 criteria have been satisfied, and therefore the Crane Travel - Spent Fuel Storage Pool LCO and Surveillances may be relocated to the TRM.

24. CTS 3/4.11 Radioactive Effluents, including: CTS 3/4.11.1.1 Concentration, CTS 3/4.11.1.2 Dose, CTS 3/4.11.1.3 Liquid Radwaste Treatment System, CTS 3/4.11.2.1 Dose Rate, CTS 3/4.11.2.2 Dose - Noble Gases, CTS 3/4.11.2.3 Dose - Iodine-131, Iodine-133, Tritium, and Radioactive Material in Particulate Form, CTS 3/4.11.2.4

Gaseous Radwaste Treatment System, CTS 3/4.11.2.5 Ventilation Exhaust Treatment System, CTS 3/4.11.2.8 Venting or Purging, CTS 3/4.11.3 Solid Radioactive Wastes, CTS 3/4.11.4 Total Dose; and 3/4.12 Radiological Environmental Monitoring, including: CTS 3/4.12.1 Monitoring Program, CTS 3/4.12.2 Land Use Census, and CTS 3/4.12.3 Interlaboratory Comparison Program.

The requirements contained within these TS are not related to any design basis accident or transient. These requirements thus do not fall within the 10 CFR 50.36 criteria, and accordingly can be relocated to the ODCM/PCP. The relocation of these requirements is in conformance with the guidance provided licensees in Generic Letter 89-01 (Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program).

The relocated specifications from the CTS discussed above are not required to be in the TS because they do not fall within the criteria for mandatory inclusion in the TS as stated in 10 CFR 50.36(c)(2)(ii). These specifications are not needed to obviate the possibility that an abnormal situation or event will give rise to an immediate threat to the public health and safety. In addition, the NRC staff has concluded that appropriate controls have been established for all of the current specifications and information that are being moved to the USAR, TRM, ODCM, PCP or ISI Program. These relocations are the subject of a new license condition discussed in Section 5.0 of this SE. Until incorporated in licensee-controlled documents, changes to these specifications and information will be controlled in accordance with the current applicable procedures and regulations that control these documents. Following implementation, the NRC may audit the removed provisions to ensure that an appropriate level of control has been achieved. The NRC staff has concluded that, in accordance with the Final Policy Statement, sufficient regulatory controls exist under the regulations, particularly 10 CFR 50.59 and 10 CFR 50.55a. Accordingly, the specifications and information, as described in detail in this SE, may be relocated from the CTS and placed in the licensee-controlled documents identified in the licensee's application dated October 16, 1998, as supplemented by letters dated December 30, 1998 (NMP2L 1844), May 10 (NMP2L 1866), June 15 (NMP2L 1872), July 30 (NMP2L 1881), August 2 (NMP2L 1883), August 11 (NMP2L 1885), August 16 (NMP2L 1886), August 19 (NMP2L 1888), August 27 (NMP2L 1893), September 10 (NMP2L 1896), September 30 (NMP2L 1900), December 14, 1999 (NMP2L 1917), and January 6, 2000 (NMP2L 1923).

F. Control of Specifications, Requirements, and Information Relocated from the CTS

In the ITS conversion, the licensee will be relocating specifications, requirements, and detailed information from the CTS to licensee-controlled documents outside the CTS. This is discussed in Sections 3.D and 3.E above. The facility and procedures described in the USAR, and TRM, which is a part of the USAR, can only be revised in accordance with the provisions of 10 CFR 50.59, which ensures records are maintained and establishes appropriate control over requirements removed from the CTS and over future changes to the requirements. Other licensee-controlled documents contain provisions for making changes consistent with applicable regulatory requirements. For example, the Offsite Dose Calculation Manual can be changed in accordance with ITS 5.5.1, and the administrative instructions that implement the QA Plan can be changed in accordance with 10 CFR 50.54(a) and 10 CFR Part 50, Appendix B. The documentation of these changes will be maintained by the licensee in accordance with the record retention requirements specified in the licensee's QA Plan for NMP2 and such applicable regulations as 10 CFR 50.59.

The license condition for the relocation of requirements from the CTS, which is discussed in Section 5.0 of this SE, will address the implementation of the ITS conversion, and the schedule for the relocation of the CTS requirements into licensee-controlled documents. The relocations to the USAR, which includes the TRM, shall be included in the next required update of this document in accordance with 10 CFR 50.71(e).

G. Evaluation of Other TS Changes (Beyond-Scope Issues) Included in the Application for Conversion to ITS

This section addresses the beyond-scope issues in which the licensee proposed changes to both the CTS and STS. The following beyond-scope issues were addressed in the notice of consideration of amendment published in the *Federal Register* on October 20, 1999 (64 FR 56518) and on December 1, 1999 (64 FR 67336).

The changes discussed below are listed in the order of the applicable ITS specification or section, as appropriate (from ITS 3.1.8 to ITS 5.5).

(1) <u>ITS 3.1.8 (DOC L.1) Scram Discharge Volume (SDV) Vent and Drain Valves</u>

The scram discharge volume (SDV) vent and drain valves primary safety function is to isolate the SDV during a scram to contain the reactor coolant discharge. The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. There are two vent and two drain valves. They close automatically on a scram signal. The isolation function can still be satisfied if at least one valve is operable in each line or the line is isolated. The licensee requested to change the SDV vent and drain valve actions to allow continued operation with one valve in each line inoperable by isolating the affected line(s) within 7 days and to allow continued operation with two valves inoperable in each line by isolating the affected line(s) within 8 hours. The 7 day allowance is also acceptable since one drain/vent valve is available during the short time for the isolation function. These increased allowances are deemed not to substantially increase the risk of a scram with an additional failure that could allow the SDV to remain unisolated; nor to substantially increase the risk of the SDV failing to accept the control rod drive water displaced during a scram. Moreover, this allowance has been previously approved for WNP-2 and LaSalle.

(2) <u>ITS 3.3.1.1 (DOC L.4) and ITS 3.3.6.1 (DOC L.6) MSLRM Scram and MSIV Closure Requirements Deleted</u>

The licensee proposed eliminating the scram and main steam line isolation valve (MSIV) closure requirements associated with the main steam line radiation monitors (MSLRM). This request in conjunction with the General Electric Licensing Topical Report NEDO-31400, and the staff's May 15, 1991, Safety Evaluation (SE) on this topical report, formed the basis for the package to be evaluated.

In the staff's SE, which accepted the referencing of NEDO-31400 for the elimination of the MSIV closure function and scram function of the MSLRM, it was stated that the following three conditions had to be met:

1) *The applicant needed to demonstrate that the assumptions with regard to input values, including power per assembly, Chi/Q (atmospheric dispersion factors), and decay times, that were made in the generic analysis, bound those for the plant.*

2) *The applicant needed to include sufficient evidence, which could be implemented or proposed operating procedures or equivalent commitments, that would provide reasonable assurance that increased significant levels of radioactivity in the main steam lines would be controlled expeditiously to limit both occupational doses and environmental releases.*

3) *The applicant needed to standardize the MSLRM and off gas radiation monitor alarm setpoint to 1.5 times the nominal N^{16} background dose rate at the monitor locations and commit to promptly sample the reactor coolant to determine possible contamination levels in the reactor coolant and the need for additional corrective action, if the MSLRM or offgas radiation monitors or both exceed their alarm setpoints.*

The licensee, in response to Condition 1 above, stated that the assumptions made in the generic analysis of the NEDO bound the NMP2 analysis. Table 1 of the submittal provides a comparison of key input parameters between the NEDO and the NMP2 Updated Safety Analysis Report (USAR) analysis assumptions. All parameters in the NEDO analysis are the same or more conservative than those in the NMP2 analysis except the power level. The effect of this one parameter on doses is more than offset by the atmospheric dispersion factors used in the NEDO analysis. The NEDO values are approximately a factor of ten greater than the NMP2 values. Table 2 of the submittal compares the control rod drop accident dose NEDO results and the NMP2 design basis. This table shows that the NEDO results are greater than those calculated by NMP2 and are well within 10 CFR Part 100 requirements. Based upon these results and the comparison of assumptions provided by the licensee, the staff finds that the licensee's analysis has met the applicable requirements of Condition 1, and is therefore acceptable.

In response to Condition 2, the licensee's submittal indicated that NMP2 has in place procedures that ensure that any significant increase in the levels of radioactivity in the main steam lines is promptly controlled. NMP2 procedure N2-SOP-17, *Fuel Failure or High Activity in Rx Coolant or Offgas* details actions to monitor main steam line and offgas system radiation levels and take reactor coolant samples to determine the extent of fuel failure. If the main steam radiation monitor alarms high, the operator is directed to isolate the source of radiation or scram the reactor to limit further releases. If the offgas radiation monitors indicate an alert or high alarm the operator is instructed to perform a power reduction. Based upon the ability of current procedures to ensure that any significant increase in the levels of radioactivity in the main steam lines and offgas system is promptly controlled to limit further releases, the staff concludes that the licensee's procedures are acceptable and responsive to Condition 2.

In response to Condition 3, NMP2 stated that the Main Steam Line Radiation Monitor alarm setpoint is 1.5 times the Nitrogen-16 background at the monitor location. The alarm will trigger entry into a procedure that will require a reactor coolant sample to be obtained and analyzed. The licensee also stated that offgas pretreatment monitor alarm/trip is set in accordance with the Offsite Dose Calculation Manual to satisfy Current Technical Specification 3.11.2.7. The Technical Specification basis for the setpoint is that by restricting the gross activity rate of noble gases from the main condenser offgas provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. Based upon a review of the licensee's commitment, the staff has determined that Condition 3 has been satisfied.

In addition to these 3 conditions NEDO-31400 addressed a condition in some plants where operating procedures allow continued bypassing of the offgas treatment system until late in the power ascension. The report states that this operating mode is acceptable, provided the offgas radiation monitors are used to automatically isolate the offgas treatment bypass line and/or offgas process line before the acceptable release rates are exceeded. NMP2 procedures allow continued bypassing of the offgas treatment system until late in the power ascension. According to the NMP2 USAR (Chapter 11.5), offgas pretreatment monitors isolate the offgas effluent upon receipt of a high radiation signal. Additionally, in the September 10, 1999 response, NMP2 stated that they have performed an evaluation of the offsite and control room doses in the event that a control rod drop accident occurs with the charcoal delay beds bypassed. This evaluation determined that the current licensing bases offsite and control room dose limits bound this scenario. Based upon the automatic isolation capability of this system during a bypass condition and the licensee's evaluation of the offgas system, the staff has determined that NMP2 meets the intent of the NEDO document.

Based on a review of the NMP2 submittal, safety analysis and information provided, the staff concludes that there are no adverse safety implications associated with elimination of the MSIV closure function and scram function of the MSLRM. The licensee has provided reasonable assurance that the radiation exposure levels are within the acceptance criterion of 10 CFR Part 50 General Design Criterion 19, Section 15.4.9 of the Standard Review Plan, and are well within 10 CFR Part 100. Therefore, the staff concludes that the proposed change to eliminate the MSIV closure function and scram function of the MSLRM is acceptable.

(3) ITS 3.3.1.1 (DOC L.10), ITS 3.5.1 (DOC L.8), ITS 3.3.6.1 (DOC L.13), ITS 3.5.2 (DOC L.5) Response Time Testing Note

A note has been added to: CTS 4.3.1.3 (proposed SR 3.3.1.1.16) for the Reactor Vessel Steam Dome Pressure — High and Reactor Vessel Water Level — Low, Level 3 Functions, CTS 4.3.2.3 (proposed SR 3.3.6.1.7 for the Main Steam Line (MSL) Isolation Reactor Vessel Water Level — Low Low Low, Level 1, Main Steam Line Pressure — Low, and Main Steam Line Flow — High Functions, and CTS 4.3.3.3 (proposed SR 3.5.1.8 and SR 3.5.2.7) the ECCS instrumentation associated with each ECCS injection/spray subsystem, that exempts the sensors or instrumentation, as applicable, from response time testing and allows the design sensor response time or instrumentation response time, as applicable, to be used in the determination of system RESPONSE TIME. Deletion of the response time test for these sensors and instrumentation was evaluated in NEDO-32291 "System Analysis for Elimination of Selected Response Time Testing Requirements," January 1994, and was determined to be acceptable provided the individual licensee referencing this NEDO in a plant specific license amendment request met several conditions stipulated in the generic SER approving NEDO-32291. The evaluation provided below is consistent with the guidance provided in the Staff's generic SER for NEDO-32291.

NMPC has performed a review of NEDO-32291 and determined that the NEDO generic analysis is applicable to NMP2. The equipment affected by the proposed change in the Technical Specifications are the RPS, ECCS, and Isolation Functions identified above. Prior to installation of a new transmitter/switch or following refurbishment of a transmitter/switch a hydraulic response time test will be performed to determine an initial sensor specific response time value. Applicable NMP2 procedures have been revised/written, as appropriate, to fulfill this recommendation. NMP2 currently does not utilize any transmitters or switches that use capillary tubes in any application that requires response time testing. Therefore, the recommendation that capillary tube testing be performed after initial installation and after any maintenance or modification activity that could damage the lines for transmitters and switches that use capillary tubes is not applicable to NMP2. Applicable calibration procedures have been revised, as appropriate, to include steps to input a fast ramp or a step change to system components during calibrations. Applicable calibration procedures have been revised, as appropriate, to assure that technicians monitor for response time degradation. In addition, technicians have received appropriate training to make them aware of the consequences of instrument response time degradation. Surveillance test procedures have been revised, as appropriate, to ensure calibrations and functional tests are being performed in a manner that allows simultaneous monitoring of both the input and output response of units under test. NMP2's compliance with the guidelines of Supplement 1 to NRC Bulletin 90-01, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount," was reviewed and documented in a safety evaluation transmitted to NMPC by NRC letter dated January 18, 1995. The NRC's evaluation concluded that NMP2's responses to Bulletin 90-01 and Supplement 1 conform to the requested actions of the Bulletin. The elimination of response time testing does not affect NMPC's response to the Bulletin. The system components for which response time testing is proposed to be eliminated has been evaluated and found to be acceptable in NEDO-32291. NMPC has reviewed the vendor recommendations for these components and confirmed that they do not contain periodic response time testing requirements.

The application of the proposed footnote will allow NMPC to use design response time data for the sensor or instrumentation, as applicable, in the determination of the system response time, and eliminate the requirement for a separate measurement of the sensor response time or instrumentation response time, as applicable. The remainder of the channel (RPS and Isolation only) will continue to be tested for response time. Other Technical Specification testing requirements such as CHANNEL CALIBRATION, CHANNEL FUNCTIONAL TEST, CHANNEL CHECK, AND LOGIC SYSTEM FUNCTIONAL TEST in conjunction with actions taken in response to NRC Bulletin 90-01 are sufficient to identify failure modes or degradations in instrument response times and assure operation of the analyzed instrument loops within acceptable limits. The elimination of the response time testing of the identified sensors and instrumentation will reduce the potential for inadvertent actuation of the system. Accordingly, this change will reduce the likelihood of a plant transient due to an inadvertent scram, isolation, or ECCS initiation. Accordingly, based on the above evaluation, which is consistent with the guidelines of the Staff's generic SER approving NEDO-32291, the proposed elimination of sensor response time and instrumentation response time is acceptable. The above change is similar to that approved by the NRC in License Amendment No. 184 for Brunswick Units 1 & 2.

(4) ITS 3.3.1.2 (DOC M.1) SNR Requirements

This change is in a more restrictive requirement that enhances safety and is therefore acceptable.

(5) ITS 3.3.2.2 (DOC L.5) Feedwater System and Main Turbine High Water Level Trip
The purpose of this instrumentation is to ensure that minimum critical power ratio (MCPR) limits are not exceeded during a feedwater controller failure, maximum demand event. This is accomplished by tripping the feedwater pumps and 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 (NMP-2 has three 50 percent feedwater pumps). The current NMP-2 TS Table 3.3.9-1 Action 140 requires a reduction in thermal power if the instrumentation is not restored to operable status. The proposed additional required action would allow removal of the associated feedwater pump from service in lieu of reducing thermal power. This required action will be used only if the instrumentation is inoperable solely due to an inoperable feedwater pump breaker. Since this proposed change accomplishes the functional purpose of the instrumentation and still ensures that a MCPR will not be exceeded , this change is acceptable.

(6) ITS 3.3.3.1 (DOC L.1), ITS 3.3.3.2 (DOC L.2), ITS 3.3.8.2 (DOC L.3), ITS 3.3.8.3 (DOC L.3), ITS 3.4.7 (DOC L.3) Six Hour Delay to Perform SR

This change adds a note to the Surveillance Requirements that will allow a 6 hour delay from entering into the associated Conditions and Required Actions for a channel placed in an inoperable status solely for performance of required Surveillances. For the Post Accident Monitoring (PAM) Instrumentation, this is only allowed provided the other channel in the associated function is operable. For the RPS Electric Power Monitoring Assemblies, this is only allowed provided the other RPS Electric Power Monitoring Assembly for the associated bus maintains trip capability. For the RCS Leakage Detection Instrumentation, this is only allowed provided the other required leakage detection instrumentation is operable. The loss of one PAM channel is acceptable in this case since another channel is operable to monitor the required function. The loss of one remote shutdown instrument channel is acceptable in this case since it does not significantly reduce the probability of monitoring the parameters, when necessary. The loss of one monitoring assembly is acceptable in this case since only one of the two assemblies is required to maintain function. The loss of one leakage detection channel is acceptable in this case since another channel is operable to monitor leakage. The short period of time (6 hours) in this condition will have no appreciable impact on risk. Also, upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to operable status or the applicable Condition must be entered and Required Actions taken. Similar 6 hour testing allowances have been granted by the NRC in TS amendments for Georgia Power Company's Hatch Unit 1 (amendment 185) and Unit 2 (amendment 125) and Washington Public Power Supply System's WNP-2 (amendment 149, the ITS amendment). This change is acceptable.

(7) ITS 3.3.4.2 (DOC L.4) Removal of Recirculation Pump Breakers from Service

CTS 3.3.4.1 Actions d and e 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 3.3.4.2 Required Action D.1, to allow removal of the associated recirculation pump breaker(s) 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, and is acceptable.

(8) ITS 3.3.4.2 (DOC M.2) Verification of ATWS Trip Time Delay and Power Level

A new Surveillance Requirement has been added (SR 3.3.4.2.4) to verify the low frequency motor generator trip portion of the Reactor Vessel Steam Dome Pressure—High Function is not bypassed for > 29 seconds when Thermal Power is > 5% RTP. This SR ensures that the Reactor Vessel Steam Dome Pressure—High Function is not inadvertently bypassed when it is required to trip the low frequency motor generators. This SR represents an additional restriction on plant operation, enhances plant safety, and therefore is acceptable.

(9) ITS 3.3.5.1 (DOCs L.11 & M.4), ITS 3.3.8.1 (DOCs L.8 & M.2), ITS 3.3.8.2 (DOCS L.4 , M.3, & M.4), ITS 3.3.8.3 (DOCs L.4, M.2 & M.3) Changes in Allowable Values & Setpoints

The licensee stated that the proposed changes are based on their most recent allowable values calculated consistent with methods described in Regulatory Guide 1.105, Revision 2, dated February 1986, ISA S 67.04-1982, and /or General Electric Setpoint Methodology described in NEDC-31336P-A, limited by the NRC SER, Revision 1, dated November 6, 1995. The proposed allowable values were established from the plant design or safety limits accounting for calibration uncertainty, process measurement uncertainty, primary element uncertainty, instrument uncertainty, and applicable environment effects. Because the proposed changes are based on allowable values calculated by the NRC approved GE methodologies above, these changes are acceptable to the staff.

(10) ITS 3.3.5.1 (DOCs L.9 & L.10) ECCS Instrumentation

Group 4 valves are the residual heat removal (RHR) sample and radioactive waste valves. The isolation function of the Group 4 isolation valves is to isolate and prevent the diversion of low-pressure coolant injection (LPCI) flow. There are only two in-series radwaste valves (2RHS*MOV142, 2RHS*MOV149) and these are in the RHR B subsystem. Also, there are four sample valves, two in -series for each of the two subsystems (2RHR*SOV 35A/B, 2RHR*SOV 36A/B). These valves need to be closed during a LOCA to prevent LPCI flow diversion. These valves are normally closed. The proposed change will require one valve in each flow path to be operable, and the valve that is required to be operable will be the associated electrically divisionalized valve. Since the sample lines are very small and the valves are normally kept closed, the proposed changes are acceptable. The licensee also proposes to delete the technical specification (TS) trip function requirements for the manual isolation pushbutton in the control room for the group 4 valves to close. The manual isolation pushbutton is not assumed in any accident or transient analysis, hence this is acceptable.

(11) ITS 3.3.5.1 (DOC L.6) ECCS Instrumentation

Group 4 valves are the RHR sample and radioactive waste valves. The isolation function of the Group 4 valves is to isolate and prevent the diversion of LPCI flow. In the current TS, Group 4 valves are treated as primary containment isolation valves (PCIVs) and therefore plant shutdown is required if the instrumentation associated with them becomes inoperable. The proposed change will remove their designation as PCIVs and therefore the actions will be specified to isolate the affected RHR line in lieu of a plant shutdown. This is acceptable.

(12) ITS 3.3.5.1 (DOC L.3) ECCS Instrumentation

The reactor pressure at which the automatic depressurization system is required to be operable is changed from 100 psig to150 psig. The low-pressure emergency core cooling system (ECCS) is capable of injecting water to the reactor for pressures well above 150 psig. Hence the proposed change is acceptable.

(13) ITS 3.3.5.1 (DOC M.6) ECCS Instrumentation

The licensee added the high-pressure core spray pump suction pressure timer to the TS. The timer is provided to preclude spurious automatic suction source swaps from the condensate storage tank to the suppression pool. Appropriate actions and surveillances have also been added. This is acceptable.

(14) ITS 3.3.6.1 (DOC L.5) Primary containment Isolation Instrumentation

The MODE 1 and 2 Applicability requirements for CTS Tables 3.3.2-1 and 4.3.2.1-1 Trip Function 1.a.3), Reactor Vessel Water Level — Low, Level 3, Trip Function 1.f, RHR Equipment Area Temperature — High, Trip Function 1.k, Reactor Building Pipe Chase Temperature — High, and Trip Function 1.l, Reactor Building Temperature — High have been deleted for the RHR SDC System (Group 5) valves. Trip Function 1.g (ITS Table 3.3.6.1-1 Function 5.c), Reactor Vessel Pressure — High, ensures that the RHR SDC System valves are isolated in MODE 1 and MODE 2 when above the RHR cut-in permissive pressure setpoint, since this Function isolates the valves when above the setpoint. When in MODE 2 below the setpoint, other Technical Specification requirements essentially ensure that RHR Shutdown Cooling is not in service (ITS 3.5.1 requires all LPCI to be OPERABLE in MODE 2, and with RHR aligned to the shutdown cooling mode, LPCI will be inoperable). In addition, plant procedures require that RHR be aligned to the LPCI mode, and the recirculation pumps to operating (which would necessitate securing the shutdown cooling mode) prior to entering MODE 2. Therefore, the deletion of MODE 1 and 2 requirements for these Functions is acceptable.

(15) ITS 3.3.6.1 (DOC A.7) Primary Containment Isolation Instrumentation, ITS 3.3.5.1 (DOC A.11) ECCS Instrumentation

Group 4 valves 2RHS*MOV142, 2RHS*MOV149, 2RHS*SOV35A/B, 2RHS*SOV36A/B are not primary containment isolation valves, hence the associated instrumentation will be moved from CTS Table 3.3.2-1 to ECCS ITS Table 3.3.5.1-1. Moving the instrumentation to the ECCS table is acceptable.

(16) ITS 3.3.8.1 (DOC L.2) DG Loss of Voltage and Degraded Voltage Channel Requirements

The current Technical Specifications require three Loss of Voltage channels and three Degraded Voltage channels for each division (even though the Minimum Channels OPERABLE column requires only two channels, Action 39 implies that three channels, as stated in the Total Number of Channels column, are required). The Division 1, 2, and Division 3 Loss of Voltage logic and Degraded Voltage logic is two-out-of-three. The instrumentation is a support system to the 4160 V ESF buses and DGs, which themselves are support systems to the various systems they provide power to. It is overly conservative to require a support system to a support system to be single failure proof. The DGs and ESF buses are designed to meet the single failure criterion, i.e., one DG and associated ESF buses are assumed to fail in the accident analyses. Therefore, ITS 3.3.8.1 only requires two Division 1, 2, and 3 Loss of Voltage channels and Degraded Voltage channels per division to be OPERABLE. A single failure of any one of these required channels will only result in the loss of one DG and associated buses, which is no worse than the loss of a single DG and associated buses for any other reason (e.g., failure of DG breaker to function properly). This is also consistent with the number required by the Minimum Channels Operable column in CTS Table 3.3.3-1, and therefore this change is acceptable.

(17) ITS 3.3.8.1 (DOC L.6) Channel Check Requirements

The CTS SR 4.3.3.1 channel check requirement for the LOP channels is not retained in the proposed NMP-2 ITS. Undervoltage relays are used to perform these functions and these relays are either in the tripped or not tripped condition, depending on the sensed voltage relative to the trip setpoint. There are no readout indication provided that can be used to compare these devices to the indications of other similar devices measuring the same parameter. The LOP channel check requirement is currently fulfilled by verifying each undervoltage relay is not tripped as indicated by the associated annunciators not providing an alarm. The current channel check provides a comparison of the tripped and not tripped status of the undervoltage relays, but does not provide indication of the overall condition of the undervoltage relays in excess of that provided by the annunciators. Thus, the verification of this status on a 12-hour frequency does not provide any additional information that is not continuously available to the plant operations staff through the absence of an actuated annunciator. However, consistent with the CTS requirement, ITS SR 3.3.8.1.1 requires LOP channel functional testing to be performed every 31 days. On these bases, this revision is acceptable.

(18) ITS 3.4.1 (DOC M.2) Stability Monitoring Power-to-Flow Verification

CTS 3.4.1.1.b requires the THERMAL POWER to be in the unrestricted zone of Figure 3.4.1.1-1. However, there is no Surveillance Requirement that verifies this requirement on a periodic basis. ITS SR 3.4.1.1 has been added to verify operation is in the "Unrestricted Zone" of ITS Figure 3.4.1-1 every 12 hours. This will ensure that entry into a region where potential instabilities can occur will not go undetected. Therefore, this change is more restrictive on plant operations and is acceptable.

(19) ITS 3.4.1 (DOC L.1) APRM and LPRM Neutron Flux Noise Value Determination

CTS 4.4.1.1.4 requires a baseline APRM and LPRM neutron flux noise value to be determined within 2 hours after entering the region for which monitoring is required. This requirement has been extended to 8 hours in the ITS, in the form of requiring the APRM and LPRM noise levels to be verified ≤ 3 times baseline noise levels within 8 hours of entering the restricted zone (ITS 3.4.1 Required Action B.1). The APRM and LPRM baseline noise levels must be known in order to perform this Required Action. The extended time to determine baseline levels the first time the region is entered after a refueling outage is consistent with the time provided in CTS 3.4.1.1 Action c to determine the APRM and LPRM noise levels are within limits. In addition, this time is acceptable since an alarm will alert the operators of a stability related power oscillation. This alarm is provided by the NUMAC Power Range Neutron Monitoring System, which has been installed during the most recent refueling outage as part of NMPC's response to Generic Letter 94-02, "Long Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors." The Technical Specification Amendment Request describing the addition of this alarm was provided in NMPC letter NMP2L-1735, dated October 31, 1997 and was approved by the NRC as documented in the NRC Letter dated April 15, 1998. The alarm is currently being tested and it is planned to be in operation prior to the implementation of the ITS.

(20) ITS 3.4.5 (DOC L.1) Surveillance Requirement Frequency Change

The Surveillance Frequency for CTS 4.4.3.2.1.b (ITS SR 3.4.5.1), has been changed from 8 hours to 12 hours, consistent with the allowance in Generic Letter 88-01, Supplement 1. The supplement allows the Frequency to be extended to once per shift, not to exceed 12 hours. NMP2 currently has a 12 hour operating shift, thus, the Frequency is adjusted to coincide with this. This is also consistent with the CTS Frequency for monitoring the airborne monitors. The staff believes that there is reasonable assurance that plant operation in this manner poses no undue risk to the health and safety of the public and therefore concludes that a surveillance frequency of 12 hours is acceptable for SR 3.4.5.1.

(21) ITS 3.5.1 (DOC L.1) ADS Valve Requirement

The number of ADS valves required to be OPERABLE in CTS 3.5.1.a and 3.5.1.b is proposed to be reduced from seven to six. CTS 3.5.1 Actions e.1 and e.2, which allow up to two of the seven ADS valves to be inoperable for a period of time prior to requiring a shutdown, and CTS 4.5.1.e.2.b), which requires each ADS valve to be opened, have also been revised to reflect this change. This change is based on the analysis summarized in Chapter 15C and in the reload analysis of Appendix A of the USAR. This analysis demonstrates adequate core cooling is provided during a small break LOCA and a simultaneous HPCS diesel generator failure (limiting LOCA) with two of the seven ADS valves out-of-service. This change reflects the credit provided through the use of NRC approved methods for calculating more realistic (yet conservative) peak cladding temperatures during accident situations. In addition, the two ADS valves out of service was approved by the NRC as documented in the initial "Safety Evaluation Report Related to the Operation of NMP2," Docket No. 50-410, Supplement No. 4 (NUREG-1047-SSER). Staff approved evaluation models were used for the analysis. The Staff concludes that the licensee requests to: (1) change the minimum number of operable ADS valves during normal power operation for ECCS function from 7 to 6 is acceptable.

(22) <u>ITS 3.5.1 (DOC L.7) ADS Valve Testing</u>

CTS 4.5.l.e.2.b requires each ADS valve to be manually opened at power at least once per 18 months. Specifically, an ADS valve disk is physically lifted by energization of an actuator solenoid, which admits nitrogen gas to a pneumatic actuator cylinder. During this test, reactor vessel steam is passed through the valve body to the suppression pool. Proposed surveillance requirement (SR) 3.5.1.7 and its Bases would permit less frequent testing of the ADS valves using an alternate approach, described below, whereby the disk is not lifted off its seat at power. The licensee proposes that each ADS actuator could be tested using either method (the current method or this alternate method).

The licensee proposes a revision to CTS 4.5.1.e.2.b to allow an alternate method of testing the ADS S/RVs. CTS 4.4.2 and CTS 4.0.5 (proposed SR 3.4.4.1 and Specification 5.5.6, respectively) require a sample population of the S/RVs to be removed and bench tested for safety-mode lift setpoint during each refueling outage to satisfy ASME Code, Section XI testing requirements. During this bench testing, the S/RVs are also stroked using the relief-mode actuator. The licensee states that safety-mode and the relief-mode bench testing of the sample population demonstrates that each installed S/RV will function properly in the safety-mode and in the relief-mode, and that the actuator of the currently installed S/RVs would successfully function. After each ADS valve is reinstalled following a bench test and after all control systems are reconnected, proposed SR 3.5.1.7 would require each ADS valve actuator to be uncoupled from its valve stem, manually actuated, and then recoupled to the valve stem. The licensee states that this proposed alternate approach verifies that the ADS controls have been properly installed prior to plant startup, without physically lifting the disk off its seat. In addition, the licensee proposes that the remaining ADS valves that have not been removed for Section XI testing during a refueling outage would be tested in a similar manner.

The staff has reviewed the licensee's proposed technical specification changes and finds that the current requirements result in opening the ADS S/RVs during power operation which could contribute to valve leakage, a stuck-open valve, the additional operation of other ECCS equipment, loss of power generation, and additional radiation exposure. The proposed alternate testing provides for actual stroking of the S/RV disks during the performance of the ASME Code setpoint testing on a sample of valves combined with stroking of the actuators after valves have been reinstalled. The staff finds that the alternate testing provided by the proposed surveillance requirement 3.5.1.7 results in the following differences between the CTS 4.5.1.e.2.b testing requirements and the proposed test requirements: (1) the proposed testing does not verify by actual stroking that the installed valve stem is properly coupled to the actuator, (2) the proposed sample testing required by the ASME Code is less frequent than the current requirement to test all ADS S/RVs each 18 months, and (3) the proposed testing does not verify, by successfully discharging the S/RVs, that the attached piping is not blocked.

The potential concern regarding the first difference is that the stem may not be properly coupled to the actuator by the proper position of the stem mounted roller bearing assembly after the S/RVs are installed and the actuators are stroked. However, the staff finds that the licensee's procedure of independently checking the repositioning of the actuator connection to the stem provides the necessary assurance of proper connection and adequately addresses this concern.

The potential concern regarding the second difference is that the S/RVs may not be adequately reliable if they are only setpoint tested and stroked at the ASME Code required frequency. The ASME Code requires these tests at a nominal frequency of once each five years with a minimum of 20% tested within 24 months. While this is significantly less frequent than the current 18-month frequency, the staff finds this acceptable since the staff has determined that meeting the Code requirement is acceptable for testing valves of this type. The ASME Code requires stroking of the S/RVs only when setpoint tests, or maintenance or repair activities, are performed.

Regarding the third difference, the licensee's foreign materials exclusion controls provide adequate assurance that no obstruction will be admitted into the S/RV discharge piping. The staff finds that this provides the necessary assurance of no obstruction in the discharge piping and is acceptable for addressing this concern.

Based on the above evaluation, the staff concludes that the proposed revised testing of the ADS S/RVs which demonstrates their depressurization function without the need for actually stroking the valve disks off the valve seats while the plant is at power, is acceptable. Therefore, the proposed SR 3.5.1.7 as a replacement for CTS 4.5.1.e.2.b is acceptable.

(23) ITS 3.6.1.2 (DOC L.5) Air Lock Door SR Frequency change

The licensee proposed to change the surveillance frequency of verifying the air lock door seal leakage rate within limits from once per 7 days when the airlock is opened for multiple entries (CTS 4.6.1.3.a.1) to once per 30 days (as described in Regulatory Guide 1.163, which is required to be met in ITS 5.5.12). This extension was recommended and approved by the NRC in Regulatory Guide 1.163, September 1995. The licensee indicated that a review of maintenance history has also shown that this test normally passes the leak rate test. This test simply confirms Operability, and the extension does not negatively impact safety. The intent of the change continues to ensure that the leakage is maintained within the proper limits, and the consequences of any analyzed event will remain bounded by the current accident analyses.

Based on the above, the staff finds the proposed change in surveillance frequency of verifying the air lock door seal leakage rate from once per 7 days to once per 30 days is acceptable as it meets the requirements of Regulatory Guide 1.163 and has a negligible effect on safety.

(24) ITS 3.6.1.3 (DOC L.9) Excess Flow Check Valve requirement to check flow is deleted

The licensee proposed to replace the requirement in CTS 4.6.3.4 that each excess flow check valve (EFCV) must check flow with the corresponding proposed ITS SR 3.6.1.3.9 that requires each EFCV to actuate to its isolation position (i.e., closed) on an actual or simulated instrument line break signal. The licensee indicated that the requirements for the EFCVs are provided in 10 CFR 50 Appendix A, GDCs 55 and 56, and in Regulatory Guide (RG) 1.11. These requirements state that there should be a high degree of assurance that the EFCVs will close or be closed if the instrument line outside containment is lost during normal reactor operation, or under accident conditions. The proposed SR ensures this requirement, since it requires each EFCV to isolate to the isolation position (closed) on an instrument line break signal. The CTS requirement does not specifically require the valve to close fully, just to "check flow". Thus, the proposed ITS SR 3.6.1.3.9 ensures the RG 1.11 requirement is met. The licensee also stated that the Instrument Line Break Analysis in the NMP2 USAR Section 15.6.2 does not even assume the valve closed. Since the actual leakage limit is not an assumption in the accident analysis, the leakage limit (i.e., check flow) is proposed to be deleted. The licensee also indicated that a similar change was approved by the NRC for the most recent BWR/5 ITS submittal.

Based on the above and an evaluation of the General Electric Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation", the staff finds the proposed change in surveillance of EFCV from check flow to actuate to their isolation position to have a negligible effect on safety and is therefore, acceptable.

(25) ITS 3.6.1.6 (DOC L.1), ITS 3.6.2.4 (DOC L.1) Spray Flows SR

Drywell spray flow (ITS 3.6.1.6)

CTS 3.6.2.2 requires the drywell spray mode of the RHR System to be capable of recirculating water from the suppression pool through the RHR heat exchangers to the drywell spray spargers. The proposed ITS 3.6.1.6 relocates the details of what constitutes an Operable drywell spray subsystem to the Bases. However, the requirement to circulate water through the heat exchanger has not been included. The licensee indicated that the drywell sprays are required to reduce pressure in the drywell and provide mixing of the atmosphere, not cool the primary containment atmosphere. These functions can be met without cooling the suppression pool water prior to spraying it into the drywell. The analysis for drywell spray does not credit cooling of the suppression pool to perform the pressure mitigation and atmosphere mixing functions. The suppression pool cooling mode, which is governed by another Technical Specification (CTS 3/4.6.2.3 and ITS 3.6.2.3) ensures heat can be removed from the primary containment, as assumed in the accident analysis. Also, the time an RHR subsystem would be in the drywell spray mode is short and this is not a concern for spray pool cooling. While the analysis for inadvertent drywell spray does credit cooling through the heat exchanger, this is to maximize the effect of the inadvertent spray. If the heat exchangers are not functioning during this event, the consequences of an inadvertent spray will not be as severe. The drywell spray system is not assumed to be an initiator of any analyzed event in which flow not through the RHR heat exchanger is more limiting. Therefore, this change does not significantly increase the consequences of an accident.

Based on the above, the staff finds the proposed change in ITS 3.6.1.6 from the CTS 3.6.2.2 for not including that the drywell spray water flow through the RHR heat exchanger is acceptable. The proposed change has a negligible effect on safety as it still provides assurance that the drywell spray system will be maintained OPERABLE, and another proposed Technical Specification will ensure cooling of the drywell to be maintained.

Suppression Pool spray flow (ITS 3.6.2.4)

CTS 3.6.2.2 and 4.6.2.2.b requires the suppression pool spray mode of the RHR System to be capable of recirculating water from the suppression pool through the RHR heat exchangers to the suppression pool spray spargers. The proposed ITS 3.6.2.4 relocates the details of what constitutes an Operable suppression pool spray subsystem to the Bases. However, the requirement to circulate water through the heat exchanger has not been included. The licensee indicated that the suppression pool sprays are required to reduce pressure in the suppression pool airspace, which will reduce pressure in the drywell. In addition, it also reduces the pressure buildup caused by bypass leakage paths. While the suppression pool spray does provide a cooling effect that also reduces pressure in the suppression pool airspace, adequate cooling effect is provided by a combination of the suppression pool sprays without flow through the RHR heat exchanger and the suppression pool cooling mode. The suppression pool cooling mode is governed by another Technical Specification (CTS 3/4.6.2.3 and ITS 3.6.2.3). The accident analysis does not credit the cooling function of the RHR heat exchangers in the pressure mitigation function of the suppression pool spray system.

Based on the above, the staff finds the proposed change in ITS 3.6.2.4 from the CTS 3.6.2.2 and 4.6.2.2 for not including that the suppression pool spray water flow through the RHR heat exchanger is acceptable. The proposed change has a negligible effect on safety as it still provides assurance that the suppression pool spray system will be maintained OPERABLE, and another proposed Technical Specification will ensure cooling of the drywell to be maintained.

(26) ITS 3.6.3.1 (DOC L.2) Hydrogen Recombiner Completion Time

CTS 3.6.6.1 ACTION only permits one hydrogen recombiner to be inoperable. If two hydrogen recombiners are inoperable CTS 3.0.3 is entered, since CTS 3.6.6.1 provides no actions for this condition. The licensee has proposed an additional ACTION in ITS 3.6.3.1(ACTION B) for the condition of both containment hydrogen recombiners inoperable. This ACTION incorporates STS 3.6.3.1 ACTION B which allows two hydrogen recombiners to be inoperable for up to 7 days provided the hydrogen control function is maintained. This new ACTION would possibly prevent unnecessary shutdown and the increased potential for transients associated with each shutdown. The use of STS 3.6.3.1 ACTION B is allowed, as specified in a Bases Reviewer's Note, provided that the alternate hydrogen control system is found to be acceptable to the staff. The licensee stated that the NMP2 nitrogen inerting and purge system can also control hydrogen in a post-LOCA environment.

The licensee indicated that the alternate hydrogen control for NMP2 has not been approved earlier since RG 1.7, Revision 2 only requires a combustible gas control system to be installed to control hydrogen. The NMP2 design includes redundant hydrogen recombiners which satisfy the requirements of RG 1.7. RG 1.7 specifically states that a containment purge system cannot be used as the primary method of controlling hydrogen after an accident but that it should be capable of aiding in cleanup. The Note in the STS Bases did not mean that ACTION B could only be used if the staff had previously accepted an alternate hydrogen control system, but that for it to be adopted as part of the ITS, the NRC needed to approve the licensee-provided alternate hydrogen control method. The NMP2 Vent and Purge System meets the RG 1.7 requirements. In combination with the inerting portion of the system, it can perform an alternate hydrogen control function (it can control hydrogen and oxygen). The NRC has previously reviewed and approved a similar method for the most recent BWR/5 ITS submittal. This method had not been previously approved for use prior to the ITS submittal, but was approved as part of the ITS submittal to be credited as a backup if both hydrogen recombiners were inoperable. This method will not be the primary method for controlling hydrogen and oxygen, but is being used to justify a 7 day Completion Time in the unlikely event that both hydrogen recombiners are inoperable.

The proposed change does not involve a significant reduction in safety. The margin of safety for this system is based on the capacity and redundancy of the system. Since the capacity is not changed and the system is backed by another method to control hydrogen, the capability for adequate response to the need for the hydrogen control function is maintained. In addition, the proposed change will prevent unnecessary shutdowns and the associated risk of potential transients.

The staff has reviewed the licensee nitrogen inerting and purge system as an alternate system to control hydrogen for a period of 7 days when the two redundant hydrogen recombiners as a primary system are not available, determined it meets RG 1.7, is similar to an earlier approved BWR/5 ITS submittal, and therefore, finds it acceptable.

(27) ITS 3.7.2 (DOC L.1) CREF Required Actions

CTS 3.7.3 Actions a and b.1 provides a 7 day restoration time when one CREF subsystem is inoperable. The CTS does not provide a restoration time when both CREF subsystems are inoperable; either LCO 3.0.3 must be entered (if in MODE 1, 2, or 3) or the CTS 3.7.3 Action b.2 must be taken (during Core Alterations, handling irradiated fuel, or OPDRVs). ITS 3.7.2 ACTION A will allow a 7 day restoration time when both CREF subsystems are inoperable, provided the CREF System safety function is maintained. ITS 3.7.2 ACTION D will require entry into 3.0.3 (if in MODE 1, 2, or 3) and ITS 3.7.2 ACTION E will require the unit to suspend Core Alterations, handling irradiated fuel, and OPDRVs (if performing one of these evolutions), if both CREF subsystems are inoperable and CREF System safety function is not maintained. The NMP2 CREF System design includes two filter trains and four air handling unit fans. For the CREF System to perform its design function, one filter train and two air handling unit fans are required. Two CREF subsystems are provided, with each subsystem consisting of one filter train and two air handling unit fans, all from the same electrical power division. Due to this design, when both subsystems are inoperable, the capability for the CREF System to perform its design function may still exist. For example, if the Division 1 filter train and the Division 2 relay room air handling unit fan are inoperable, sufficient components are OPERABLE for the CREF System to meet its safety function (using the Division 2 filter train, the Division 1 relay room air handling unit fan, and either the Division 1 or 2 control room area air handling unit fan). Therefore, since this alignment is equivalent to having one CREF subsystem fully OPERABLE, the 7 day restoration time is acceptable, provided the CREF System safety function is maintained. The 7 day restoration time is identical to that already allowed in the CTS when one CREF subsystem is inoperable. In the current condition allowed by the CTS, the remaining OPERABLE subsystem will perform the CREF System safety function, assuming no additional single failure. The proposed condition will still ensure the remaining OPERABLE components of the two subsystems can perform the CREF safety function, assuming no additional single failure. If the remaining components of the CREF subsystems cannot maintain the CREF System safety function, then the unit will be required to enter LCO 3.0.3 (if in MODE 1, 2, or 3), or the unit must suspend Core Alterations, handling irradiated fuel, and OPDRVs (if performing one of these evolutions), consistent with the current requirements. In addition, this concept is consistent with the ECCS Specification in NUREG-1430, NUREG-1431, and NUREG-1432, which allow multiple ECCS trains to be inoperable for the same length of time as is currently allowed for one train only, provided 100% of the flow equivalent to a single ECCS train is available.

Due to this change, CTS 3.7.3 Action b.1 (ITS 3.7.2 Required Action C.1) has been revised to require the Operable components of CREF subsystem(s) equivalent to a single CREF subsystem to be placed in operation in lieu of placing the Operable subsystem in operation. The purpose of the current Action to place the subsystem in operation, is to ensure that the remaining subsystem is Operable, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected. Since this change does not impact the purpose of the Action (the three listed reasons remain valid), this change is acceptable.

(28) <u>ITS 3.7.3 (DOC L.4) Control Room Envelope AC System</u>

CTS 3.7.3 Actions a and b.1 provides a 7 day restoration time when one control room envelope AC subsystem is inoperable. The CTS does not provide a restoration time when both control room envelope AC subsystems are inoperable; either LCO 3.0.3 must be entered (if in MODE 1, 2, or 3) or the CTS 3.7.3 Action b.2 must be taken (during Core Alterations, handling irradiated fuel, or OPDRVs). ITS 3.7.3 ACTION A will allow a 30 day restoration time when both control room envelope AC subsystems are inoperable, provided the Control Room Envelope AC System safety function is maintained. ITS 3.7.3 ACTION D will require entry into 3.0.3 (if in MODE 1, 2, or 3) and ITS 3.7.3 ACTION E will require the unit to suspend Core Alterations, handling irradiated fuel, and OPDRVs (if performing one of these evolutions), if both control room envelope AC subsystems are inoperable and Control Room Envelope AC System safety function is not maintained. The NMP2 Control Room Envelope AC System design includes four air handling units. For the Control Room Envelope AC System to perform its design function, two air handling units are required. Two Control Room Envelope AC subsystems are provided with each system including two air handling units, both from the same electrical power division. Due to this design, when both subsystems are inoperable, the capability for the Control Room Envelope AC System to perform its design function may still exist. For example, if the Division 1 control room area and the Division 2 relay room air handling units are inoperable, sufficient components are operable for the Control Room Envelope AC System to meet its safety function (using the Division 2 control room area air handling unit and the Division 1 relay room air handling unit). Since this alignment is equivalent to having one control room envelope AC subsystem fully OPERABLE, the 30 day restoration time is acceptable, provided the Control Room Envelope AC System safety function is maintained. The 30 day restoration time is additional single failure. If the remaining components of the control room envelope AC subsystems cannot maintain the Control Room Envelope AC System safety function, then the unit will be required to enter LCO 3.0.3 (if in MODE 1, 2, or 3), or the unit must suspend Core Alterations, handling irradiated fuel, and OPDRVs (if performing one of these evolutions), consistent with the current requirements. In addition, this concept is consistent with the ECCS Specification in NUREG-1430, NUREG-1431, and NUREG-1432, which allow multiple ECCS trains to be inoperable for the same length of time as is currently allowed for one train only, provided 100% of the flow equivalent to a single ECCS train is available.

Due to this change, CTS 3.7.3 Action b.1 (ITS 3.7.3 Required Action C.1) has been revised to require the Operable components of control room envelope AC subsystem(s) equivalent to a single control room envelope AC subsystem to be placed in operation in lieu of placing the Operable subsystem in operation. The purpose of the current Action to place the subsystem in operation is to ensure that the remaining subsystem is Operable, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected. Since this change does not impact the purpose of the Action (the three listed reasons remain valid), this change is acceptable.

(29) ITS 3.8.1 (DOC L.9) Revising the Required Loading Kilowatt Values for the 24-Hour Emergency Diesel Generator Surveillance

The CTS 24-hour EDG SR 4.8.1.1.2.e.8 requires that the Divisions I and II EDGs operate loaded to ≥4840 kW for the first 2 hours of the test and to ≥4400 kW for the remaining 22 hours. This CTS SR also requires that the Division III EDG operate loaded to ≥2860 kW for the first 2 hours of the test and to ≥2600 kW for the remaining 22 hours. The corresponding proposed ITS SR 3.8.1.12 requires that the Divisions I and II EDGs operate with a load of ≥4620 kW and ≤4840 kW for ≥2 hours of the test and with a load of ≥3960 kW and ≤4400 kW for the remaining hours of the 24-hour test. This ITS SR also requires that the Division III EDG operate with a load of ≥2730 kW and ≤2860 kW for ≥2 hours of the test and with a load of ≥2340 kW and ≤2600 kW for the remaining hours of the 24-hour test.

The proposed ITS SR 3.8.1.12 relaxes the loading requirements for the EDG 24 hour surveillance test. The revised 22 hour loading requirement for each EDG is 90% - 100% of the continuous rating of the EDG. The revised 2 hour loading requirement for each EDG is 105% - 110% of the continuous rating of the EDG. These new proposed load range values for the EDG 24 hour tests are consistent with the recommendations contained in Regulatory Guide 1.9, Revision 3, "Selection, Design, Qualification, And Testing Of Emergency Diesel Generator Units Used As Class 1E Onsite Electric Power Systems At Nuclear Power Plants." These new load range values preclude routine overloading of the EDGs while the lower values provide assurance that the EDGs are at operating temperatures. In addition, the proposed ITS SR continues to provide assurance that the EDGs will carry normal and rated loads, and therefore is acceptable.

(30) ITS 3.8.1 (DOC L.15) Revision of the Currently Required Time for the Emergency Diesel Generators to Start and Energize the Emergency Buses From a Loss of Voltage Signal

The CTS SRs 4.8.1.1.2.e.4.a) 2) and 4.8.1.1.2.e.4.b) 2) require the EDGs to start and energize the emergency buses within 13 seconds of a loss of offsite power signal. The proposed corresponding NMP-2 ITS SR 3.8.1.9 requires the EDGs to start and energize the emergency buses within 13.20 seconds. This proposed time is the summation of the current EDG start time of 10 seconds from various CTS Section 4.8.1.1 surveillances and the EDG loss of voltage time delay allowable value from the CTS Table 3.3.3-2. This is also the time assumed in the accident analysis for the EDG to start when only a loss of voltage occurs. The time of 13 seconds provided in the CTS SRs is the allowed EDG start and emergency bus energization time rounded to the nearest whole second. Hence, this revision makes the EDG start and emergency bus energization time required in CTS SRs 4.8.1.1.2.e.4.a)2) and 4.8.1.1.2.e.4.b)2) consistent with the currently allowed times provided in other portions of the CTS. This revision is therefore an administrative item and is acceptable.

(31) <u>ITS 3.8.2 (DOC L.4) Addition of a Note Which Exempts Surveillances Pertaining to Emergency Diesel Generator Starting on a LOCA Signal and a LOCA/LOOP Signal While in Modes 4 and 5, and During Handling of Irradiated Fuel in the Secondary Containment</u>

CTS 4.8.1.2, which provides the surveillance requirements for alternating current sources while in Modes 4 and 5 and during handling of irradiated fuel in the secondary containment, requires the SRs of CTS 4.8.1.1.2 to be performed. Two of these surveillances are the EDG start on an ECCS initiation signal and the EDG start and load on an ECCS initiation signal concurrent with a loss of offsite power signal. A note has been provided with the proposed NMP-2 ITS SR 3.8.2.1 which exempts these two surveillances when the associated ECCS subsystems are not required to be operable. The CTS and the NMP-2 ITS do not require the ECCS subsystems to be operable in Mode 5 when the spent fuel storage pool gates are removed and water level is ≥22 feet 3 inches above the top of the reactor pressure vessel flange. The CTS and the ITS also do not require the ECCS subsystems to be operable when defueled. The EDGs are required to support the equipment powered from the emergency buses. However, when the ECCS subsystems are not required to be operable, then there is no technical reason to require the EDG to autostart on an ECCS initiation signal since this results in a requirement for the support system to be operable when the supported system is not required to be operable. In addition, the ECCS initiation signal is an anticipatory start signal for the EDGs which are only needed during a LOCA if a loss of offsite power occurs. The requirement to autostart required EDGs on a loss of offsite power signal is retained in the proposed ITS SR 3.8.1.9. Thus, when in Modes 4 and 5 and during handling of irradiated fuel in the secondary containment, when the associated ECCS subsystems are not required to be operable, there is no reason to require the EDGs to be capable of automatically starting on an ECCS actuation signal either alone or concurrent with a loss of offsite power signal. On this basis, this revision is acceptable.

(32) <u>ITS 3.8.3 (DOC M.2)Increasing the Fuel Oil Storage Tank Limits for the Divisions I and II Emergency Diesel Generators as Well as the Six Day Limits for All Emergency Diesel Generators</u>

The CTS LCOs 3.8.1.1.a.2 and 3.8.1.2.b.2 require the level of the Divisions I and II EDG fuel oil storage tanks to be ≥47,547 gallons. In addition, CTS 3.8.1.1 Action j and 3.8.1.2 Action d require the 6-day fuel oil storage tank levels to be ≥40,755 gallons for Division I and II EDGs and ≥30,293 gallons for the Division III EDG. The level for Divisions I and II EDG fuel oil storage tanks is increased in proposed ITS SR 3.8.3.1 to ≥50,000 gallons and the 6 day levels are increased in ITS 3.8.3 Condition A to ≥44,000 gallons for Divisions I and II fuel oil storage tanks and to ≥30,813 gallons for the Division III EDG storage tank. These proposed values are based on the most recent calculations, which increase the amount of fuel oil needed. These values provide assurance the EDGs have sufficient fuel oil to operate for the assumed 7 days and 6 days, respectively. These revised proposed EDG fuel oil storage tank limits are more restrictive on plant operations and provide additional requirements not include in the CTS and are acceptable.

(33) <u>ITS 3.8.4 (DOC L.3) Revision of the Battery Load Profile to be Consistent With the Load Profile Specified in the Updated Safety Analysis Report</u>

CTS SR 4.8.2.1.d.2 requires battery service test capacity to be adequate to supply specific dummy load current profiles while maintaining the battery terminal voltage ≥105 volts for Divisions I and II and ≥112.5 volts for Division III. For the Division I battery, the CTS require a dummy load current profile of ≥818 amperes during the initial 60 seconds; ≥445 amperes during the next 118 minutes; and ≥701 amperes during the remainder of the 2-hour test. For the Division II battery, the CTS require a dummy load current of ≥570 amperes during the initial 60 seconds; ≥449 amperes during the next 118 minutes; and ≥505 amperes during the remainder of the 2-hour test. For the Division III battery, the CTS require a dummy load current of ≥54.6 amperes during the initial 60 seconds and ≥15.4 amperes during the remainder of the 2-hour test. The proposed ITS revises the required battery service test dummy load current profile for each battery and proposes to relocate the revised battery service test current profiles to the USAR. From the markup copy of the CTS for the Division I battery, the ITS require a dummy load current of ≥721 amperes during the initial 60 seconds; ≥234 amperes during the next 118 minutes; and ≥570 amperes for the remainder of the 2-hour test. From the markup copy of the CTS for the Division II battery, the ITS require a dummy load current of ≥301 amperes during the initial 60 seconds; ≥193 amperes during the next 118 minutes; and ≥223 amperes during the remainder of the 2-hour test. From the markup copy of the CTS for the Division III battery, the ITS require a dummy load current ≥47.9 amperes during the initial 60 seconds and ≥15.4 amperes during the remainder of the 2-hour test.

CTS SR 4.8.2.1.d.2 requires specific current rates to be used for battery service testing with a dummy load. The CTS required dummy load current profiles were consistent with the load profiles provided in the USAR Tables 8.3-8, 8.3-9, and 8.3-10 for Divisions I, II, and III, respectively, at the time of issuance of the low power operating license for NMP-2. The load current profiles currently provided in these USAR tables were revised since the issuance of the low power license for NMP-2. The proposed NMP-2 ITS SR 3.8.4.7 does not specify dummy load current profile rates for battery service testing, but the above current profile rates are consistent with the battery load current profiles currently provided in the USAR Tables 8.3-8, 8.3-9, and 8.3-10. The service test dummy load current profiles are reduced for each of the consistent with the battery load current profiles currently provided in the USAR Tables 8.3-8, 8.3-9, and 8.3-10. The service test dummy load current profiles are reduced for each of the three divisional batteries for every period of duty cycle with the exception of one duty cycle for the Division III battery. Specifically, the 1 to 120 minute duty cycle for the Division III battery remains unchanged at 15.4 amperes. The service test dummy load current profiles are reduced to be consistent with actual emergency loads and the as-built condition of the plant. However, the revised service test load current profiles continue to include all necessary loads to support the operation of safety-related equipment under design basis accident LOCA conditions. With the revised required current profiles, the spare capacity is increased for each of the divisional batteries by decreasing the demand on each battery. Thus, the proposed ITS revised requirement for battery service testing is acceptable since it continues to demonstrate the batteries have capability to supply power to all required safety loads consistent with design requirements.

(34) ITS 3.8.4 (DOC L.4) Addition of an Allowance to Perform a Battery Modified Performance Discharge Test Every Cycle in Lieu of a Battery Service Test

CTS SR 4.8.2.1.e allows a battery performance discharge test to substitute for the battery service test once every 60 months. Note 1 provided with the proposed ITS SR 3.8.4.7 allows a modified performance discharge test to be substituted for the battery service test. In addition, the modified performance discharge test is allowed to be substituted for the battery service test at any time. The modified performance discharge test normally consists of a 3 hour test with two rates. The one minute rate published for the battery or the largest current load of the duty cycle followed by the test rate used for the performance discharge test. This test may consist of a single current rate if the test current rate used for the performance discharge test exceeds the one minute current rate. The service test consists of a two-hour duty cycle with two or three current rates, depending on the battery being tested. The one minute rate for the largest current load of the duty cycle, the current rate based on the steady state loads of the duty cycle from 1 minute to 119 minutes, and a final one minute current rate based on the cycling loads of the duty cycle. For the NMP-2 plant, the second test current rate to be used for the modified performance discharge test is greater than both the steady state and the cycling loads (1 minute through 120 minutes) of the service test. Thus, for the NMP-2 plant, the modified performance discharge test is a more severe test of the battery than the service test. To assure that the modified performance discharge test can only be used as a substitute as long as it remains a more severe test of the battery, Note 1 of the proposed ITS SR 3.8.4.7 only allows the substitution as long as the modified performance discharge test current completely envelops the service test current. The proposed revision also permits the NMP-2 licensee to perform the modified performance discharge test every refueling outage, in lieu of the service test. Performing the modified performance discharge test, every refueling outage allows for better trending of battery capacity with more data points over an expected 20-year battery service life. At the same time, the service use of the battery is continuing to be verified. This will also allow the licensee to more accurately identify when the battery is approaching degradation so that corrective action can be taken in a timely manner. The additional deep discharges that will result from performing the modified performance discharge test more frequently will not significantly affect the batteries since each battery is designed for 30 such discharges. Performing a modified performance discharge test every 24-months results in 10 deep discharge cycles over an expected 20-year battery service life and thus 20 such cycles remain for any plant DC (direct current) system challenges. On this technical basis, the ITS proposed requirement revisions are acceptable.

(35) ITS 3.8.7 (DOC M.1) Requiring the Inverters to be Powered From an Uninterruptible Power Source (Direct Current Source)

The CTS LCOs 3.8.3.1.a.1.c and 3.8.3.1.a.2.c require the energization of the Divisions I and II 120-volt AC (alternating current) distribution panels be from the inverters identified as 2VBA*UPS2A and 2VBA*UPS2B, respectively. A footnote for these CTS LCOs requires the inverters to be energized from their normal AC supply or their backup DC (Direct Current) supply. The proposed NMP-2 ITS 3.8.7, revises the footnote requirement to clearly define that an operable inverter is one that has the capability of being supplied without interruption from its associated DC source, which is consistent with how the CTS requirement is implemented by plant procedures. Each inverter is normally supplied by its associated normal AC source. If the normal AC source is not available, the DC source is designed to supply the associated inverter and the inverter may be considered operable. This provides assurance that the inverter is capable of supplying the loads without interruption. An uninterruptible supply is required to support the design basis accidents to assure proper emergency core cooling system operation. In addition, the inverters are also required to support other technical specifications equipment such as the reactor core isolation cooling system. Since the words in the CTS footnote do not specifically require uninterruptible power supplies (that is, either the AC or the DC supply can be used to energize an inverter), this ITS revision is a more restrictive requirement on plant operations, and is acceptable.

(36) ITS 5.5.2.b (DOC A.2) Addition of SR Note

This is an administrative change, consistent with the current practices, that has no adverse effect upon safety and is therefore acceptable.

4.0 COMMITMENTS RELIED UPON

In reviewing the proposed ITS conversion for NMP2, the staff has relied upon the licensee commitment to relocate certain requirements from the CTS to licensee-controlled documents as described in Table R, "Relocated Specifications and Removal of Details Matrix," attached to this SE. This table reflects the relocations described in the licensee's submittals on the conversion. The staff requested and the licensee submitted a license condition to make this commitment enforceable (see Section 5.0). Such a commitment from the licensee is important to the ITS conversion because the acceptability of removing certain requirements from the TS is based on those requirements being relocated to licensee-controlled documents where further changes to the requirements will be controlled by regulations or other requirements (e.g., in accordance with 10 CFR 50.59).

5.0 LICENSE CONDITIONS

A license condition to define the schedule to begin performing the new and revised SRs after the implementation of the ITS is to be included in the license amendment issuing the ITS. This schedule is:

- For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.

- For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.

- For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.

- For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.

The staff has reviewed the above schedule for the licensee to begin performing the new and revised SRs and concludes that it is an acceptable schedule.

Also, a license condition is to be included that will enforce the relocation of requirements from the CTS to licensee-controlled documents. The relocations are provided in Table R, "Relocated Specifications and Removal of Details Matrix," and Section 3.E above, "Relocated Entire CTS Specifications." The license condition states that the relocations would be completed no later than August 31, 2000, and the relocations to the USAR shall be reflected in updates completed in accordance with 10 CFR 50.71(e). This schedule is acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the ITS conversion amendment for NMP2. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on December 15, 1999 (64 FR 70073), for the proposed conversion of the CTS to ITS for NMP2. Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

With respect to other changes included in the application for conversion to Improve Technical Specifications the items change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments required by these other changes involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. In two sets of notices, the Commission issued proposed findings that the amendments required by these other changes involve no significant hazards consideration, and there has been no public comment on these findings published at: (a) 64 FR 56518 (October 20, 1999) and (b) 64 FR 67336 (December 1, 1999). Accordingly, these changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the implementation of these changes.

8.0 CONCLUSION

The NMP2 ITS provides clearer, more readily understandable requirements to ensure safe operation of the plant. The NRC staff concludes that the ITS for NMP2 satisfy the guidance in the Final Policy Statement on TS improvements for nuclear power reactors with regard to the content of TS, and conform to the STS provided in NUREG-1433, Revision 1, or NUREG-1434, Revision 1, with appropriate modifications for plant-specific considerations. The NRC staff further concludes that the ITS satisfy Section 182a of the Atomic Energy Act, 10 CFR 50.36, and other applicable standards. On this basis, the NRC staff concludes that the proposed ITS for NMP2 are acceptable.

The NRC staff has also reviewed the plant-specific changes to the CTS as described in this SE. On the basis of the evaluations described herein for each of the changes, the NRC staff also concludes that these changes are acceptable.

The Commission has concluded, based on the considerations discussed above, that:
(1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security, or to the health and safety of the public.

Attachments:
1. List of Acronyms
2. Table A of Administrative Changes to Current Technical Specifications
3. Table M of More Restrictive Changes to Current Technical Specifications
4. Table L of Less Restrictive Change to Current Technical Specifications
5. Table R of Relocated Specifications and Removed Details from Current Technical Specifications

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List of Acronyms

AC	Air Conditioning or Alternating Current
ADS	Automatic Depressurization System
AOT	Allowed Outage Time
APLHGR	Average Planar Linear Heat Generation Rate
APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
ATWS-RPT	Anticipated Transient Without Scram - Recirculation Pump Trip
BPWS	Banked Position Withdrawal Sequence
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CFR	Code of Federal Regulations
CFT	Channel Functional Test
COLR	Core Operating Limits Report
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CREF	Control Room Envelope Filtration
CST	Condensate Storage Tank
CTS	Current Technical Specification
DBA	Design-Basis Accident
DC	Direct Current
DG	Diesel Generator
DOC	Discussion of Change (from the CTS)
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFCV	Excess Flow Check Valve
EOC-RPT	End of Cycle - Recirculation Pump Trip
EPA	Electrical Protection Assembly
ESF	Engineered Safeguard Feature
FR	Federal Register
FRTP	Fraction of Rated Thermal Power
GDC	General Design Criteria
GE	General Electric
HEPA	High Efficiency Particulate Air
HPCS	High Pressure Core Spray
Hz	Hertz
IRM	Intermediate Range Monitor
ISI	Inservice Inspection
ITS	Improved (converted) Technical Specifications
Kv	Kilovolt
kW	Kilowatt
LCO	Limiting Condition for Operation
LHGR	Linear Heat Generation Rate
LLS	Low-Low Set
LOCA	Loss of Coolant Accident

ATTACHMENT 1

LOOP	Loss of Offsite Power
LOP	Loss of Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
LSFT	Logic System Functional Test
MCPR	Minimum Critical Power Ratio
MFLPD	Maximum Fraction of Limiting Power Density
MG	Motor Generator
MSIV	Main Steam Isolation Valve
MWD/T	Megawatt Days/short Ton
NMP2	Nine Mile Point Unit 2
NUMAC	Nuclear Measurement Analysis and Control
OPDRV	Operation with a Potential for Draining the Reactor Vessel
PAM	Post-Accident Monitoring
P/T	Pressure/Temperature
QA	Quality Assurance
RAI	Request for Additional Information
RBM	Rod Block Monitor
RCS	Reactor Coolant System
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSCS	Rod Sequence Control System
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWM	Rod Worth Minimizer
SCIV	Secondary Containment Isolation Valve
SDC	Shutdown Cooling
SDM	Shutdown Margin
SDV	Scram Discharge Volume
SE	Safety Evaluation
SER	Safety Evaluation Report
SGT	Standby Gas Treatment
SLC	Standby Liquid Control
SR	Surveillance Requirement
SRM	Source Range Monitor
SRV	Safety/Relief Valve
SSER	Supplemental Safety Evaluation Report
STS	Improved Standard Technical Specification(s), NUREG-1433/4, Rev. 1
SW	Service Water
TRM	Technical Requirements Manual
TS	Technical Specifications
TSTF	Technical Specifications Task Force (re: generic changes to the STS)
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply

USAR	Updated Final Safety Analysis Report
V	Volt
VAC	Volts Alternating Current

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR

NMP2

TABLE OF CONTENTS

(continued)

(continued)

(continued)

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

(continued)

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, 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.
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 conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites;" Table E-7 of

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial movement of the associated turbine stop valves or turbine control valves to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

(continued)

1.1 Definitions

LEAKAGE (continued)	2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. <u>Unidentified LEAKAGE</u> All LEAKAGE into the drywell that is not identified LEAKAGE; and c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
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 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 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.

(continued)

1.1 Definitions (continued)

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. These tests are: a. Described in Chapter 14, Initial Test Program of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3467 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

1.1 Definitions (continued)

SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components: a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[(a)] or Startup/Hot Standby	NA
3	Hot Shutdown[(a)]	Shutdown	> 200
4	Cold Shutdown[(a)]	Shutdown	≤ 200
5	Refueling[(b)]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.
EXAMPLES	The following examples illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example, the logical connector AND is used to indicate that, when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition. However, when a <u>subsequent</u> division, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

(continued)

1.3 Completion Times

DESCRIPTION (continued)

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extension does not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each division, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 12 hours AND in MODE 4 within 36 hours. A total of 12 hours is allowed for reaching MODE 3 and a total of 36 hours (not 48 hours) is allowed for reaching MODE 4 from the time that Condition B was entered. If MODE 3 is reached within 6 hours, the time allowed for reaching MODE 4 is the next 30 hours because the total time allowed for reaching MODE 4 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 4 is the next 36 hours.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X subsystem inoperable.	A.1 Restore Function X subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y subsystem inoperable.	B.1 Restore Function Y subsystem to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X subsystem inoperable. AND One Function Y subsystem inoperable.	C.1 Restore Function X subsystem to OPERABLE status. OR C.2 Restore Function Y subsystem to OPERABLE status.	72 hours 72 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X subsystem and one Function Y subsystem are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each subsystem, starting from the time each subsystem was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second subsystem was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected subsystem was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock". In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

---------------------------NOTE---------------------------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be completed within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1

(continued)

1.3 Completion Times

EXAMPLES	EXAMPLE 1.3-7 (continued) is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements. Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. Example 1.4-4 discusses these special situations. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction. The use of "met" or "performed" in these instances conveys specified meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria. SR 3.0.4 restrictions would not apply if both the following conditions are satisfied:

(continued)

1.4 Frequency

DESCRIPTION (continued)

a. The Surveillance is not required to be performed; and

b. The Surveillance is not required to be met or, even if required to be met, is not known to be failed.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Examples 1.4-3 and 1.4-4), then SR 3.0.3 becomes applicable.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-1 (continued)

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Not required to be performed until 12 hours after ≥ 25% RTP. -- Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

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

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

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

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.
LCO 3.0.3	When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 2 within 7 hours; b. MODE 3 within 13 hours; and c. MODE 4 within 37 hours. Exceptions to this Specification are stated in the individual Specifications. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, and 3.
LCO 3.0.4	When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)	Exceptions to this Specification are stated in the individual Specifications. LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3.
LCO 3.0.5	Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.
LCO 3.0.6	When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.
LCO 3.0.7	Special Operations LCOs in Section 3.10 allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Special Operations LCOs is optional. When a Special Operations LCO is desired to be met but is not met, the ACTIONS of the Special Operations LCO shall be met. When a Special Operations LCO is not

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.7 (continued)	desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

3.0 SR APPLICABILITY (continued)

SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be:

a. $\geq$ 0.38% $\Delta k/k$, with the highest worth control rod analytically determined; or

b. $\geq$ 0.28% $\Delta k/k$, with the highest worth control rod determined by test.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SDM not within limits in MODE 1 or 2.	A.1	Restore SDM to within limits.	6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
C. SDM not within limits in MODE 3.	C.1	Initiate action to fully insert all insertable control rods.	Immediately
D. SDM not within limits in MODE 4.	D.1	Initiate action to fully insert all insertable control rods.	Immediately
	<u>AND</u>		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	<u>AND</u>		
	D.3	Initiate action to restore one standby gas treatment (SGT) subsystem to OPERABLE status.	1 hour
	<u>AND</u>		
	D.4	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	<u>AND</u>		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	<u>AND</u>		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.3	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	<u>AND</u>		
	E.4	Initiate action to restore one SGT subsystem to OPERABLE status.	1 hour
	<u>AND</u>		
	E.5	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.1.1	Verify SDM is: a. $\geq$ 0.38% $\Delta k/k$ with the highest worth control rod analytically determined; or b. $\geq$ 0.28% $\Delta k/k$ with the highest worth control rod determined by test.	Prior to each in vessel fuel movement during fuel loading sequence AND Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

LCO 3.1.2 The reactivity difference between the monitored rod density and the predicted rod density shall be within ± 1% $\Delta k/k$.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity difference not within limit.	A.1 Restore core reactivity difference to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the monitored rod density and the predicted rod density is within ± 1% $\Delta k/k$.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement <u>AND</u> 1000 MWD/T thereafter during operations in MODE 1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	------------NOTE------------ Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. --------------------------------	
	A.1 Verify stuck control rod separation criteria are met.	Immediately
	AND	
	A.2 Disarm the associated control rod drive (CRD).	2 hours
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	AND		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	--------NOTE--------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. -------------------- Fully insert inoperable control rod.	3 hours
	AND		
	C.2	Disarm the associated CRD.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ---------NOTE--------- Not applicable when THERMAL POWER > 10% RTP. ---------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1 Restore compliance with BPWS. OR D.2 Restore control rod to OPERABLE status.	4 hours 4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. OR Nine or more control rods inoperable.	E.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.3.1 Determine the position of each control rod.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	-------------------NOTE------------------- Not required to be performed until 7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. --- Insert each fully withdrawn control rod at least one notch.	7 days
SR 3.1.3.3	-------------------NOTE------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. --- Insert each partially withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 05 is $\leq$ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.5	Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position AND Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	120 days cumulative operation in MODE 1
SR 3.1.4.3	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after fuel movement within the affected core cell <u>AND</u> Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1
Control Rod Scram Times

-----NOTES-----

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

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

NOTCH POSITION	SCRAM TIMES[a][b] (seconds) WHEN REACTOR STEAM DOME PRESSURE $\geq$ 800 psig
45	0.528
39	0.866
25	1.917
05	3.437

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

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ 900 psig.	A.1 --------NOTE-------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. --------------------	
	Declare the associated control rod scram time "slow."	8 hours
	<u>OR</u>	
	A.2 Declare the associated control rod inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig.	B.1	Restore charging water header pressure to ≥ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
	<u>AND</u>		
	B.2.1	--------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. --------------------- Declare the associated control rod scram time "slow."	1 hour
	<u>OR</u>		
	B.2.2	Declare the associated control rod inoperable.	1 hour
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig.	C.1	Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig
	<u>AND</u>		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2	Declare the associated control rod inoperable.	1 hour
D. Required Action B.1 or C.1 and associated Completion Time not met.	D.1	--------NOTE--------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. --------------------- Place the reactor mode switch in the shutdown position.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each control rod scram accumulator pressure is ≥ 940 psig.	7 days

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS).

APPLICABILITY: MODES 1 and 2 with THERMAL POWER ≤ 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more OPERABLE control rods not in compliance with BPWS.	A.1 --------NOTE--------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." --------------------- Move associated control rod(s) to correct position.	8 hours
	OR	
	A.2 Declare associated control rod(s) inoperable.	8 hours

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify all OPERABLE control rods comply with BPWS.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is ≥ 70°F.	24 hours
SR 3.1.7.3	Verify temperature of pump suction piping up to the pump suction valve is ≥ 70°F.	24 hours
SR 3.1.7.4	Verify continuity of explosive charge.	31 days
SR 3.1.7.5	Verify the concentration of sodium pentaborate in solution is within the limits of Figure 3.1.7-1.	31 days AND Once within 24 hours after water or sodium pentaborate is added to solution AND Once within 24 hours after solution temperature is restored to ≥ 70°F
SR 3.1.7.6	Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1235 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction valve is unblocked.	24 months <u>AND</u> Once within 24 hours after piping temperature is restored to ≥ 70°F

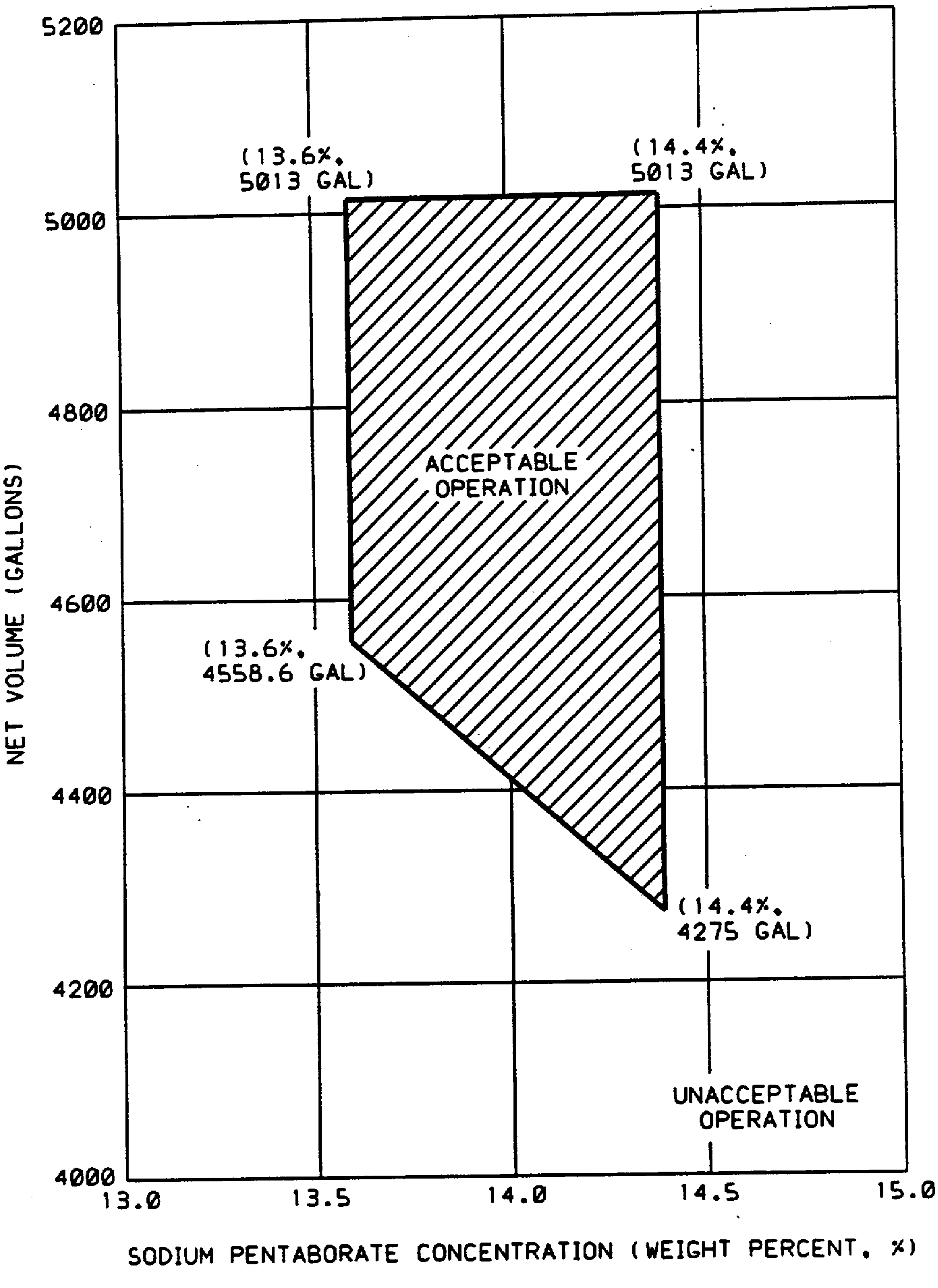

Figure 3.1.7-1 (Page 1 of 1)
Sodium Pentaborate Solution Volume/Concentration Requirements

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

LCO 3.1.8 Each SDV vent and drain valve shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

1. Separate Condition entry is allowed for each SDV vent and drain line.

2. An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SDV vent or drain lines with one valve inoperable.	A.1 Isolate the associated line.	7 days
B. One or more SDV vent or drain lines with both valves inoperable.	B.1 Isolate the associated line.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	-------------------NOTE------------------- Not required to be met on vent and drain valves closed during performance of SR 3.1.8.2. -- Verify each SDV vent and drain valve is open.	31 days
SR 3.1.8.2	Cycle each SDV vent and drain valve to the fully closed and fully open position.	92 days
SR 3.1.8.3	Verify each SDV vent and drain valve: a. Closes in ≤ 30 seconds after receipt of an actual or simulated scram signal; and b. Opens when the actual or simulated scram signal is reset.	24 months

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP <u>AND</u> 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP <u>AND</u> 24 hours thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

LCO 3.2.4

a. MFLPD shall be less than or equal to Fraction of RTP (FRTP); or

b. Each required APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value shall be modified by ≤ FRTP/MFLPD; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times MFLPD.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	-------------------NOTE------------------- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements. --- Verify MFLPD is within limits.	Once within 12 hours after ≥ 25% RTP <u>AND</u> 24 hours thereafter
SR 3.2.4.2	-------------------NOTE------------------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements. --- Verify each required: a. APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value is modified by ≤ FRTP/MFLPD; or b. APRM gain is adjusted such that the APRM reading is ≥ 100% times MFLPD.	12 hours

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

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 Place associated trip system in trip.	12 hours 12 hours
B. --------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, and 2.d. -------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS 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 RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.3	-----NOTE----- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. ----- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.4	-----NOTE----- For Functions 1.a and 1.b, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ----- Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to fully withdrawing SRMs
SR 3.3.1.1.6	-----------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. --- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 effective full power hours
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	Calibrate the trip units.	92 days
SR 3.3.1.1.10	-----------------NOTE------------------ For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.11	Perform CHANNEL CALIBRATION.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	-------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Functions 1.a and 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Verify Turbine Stop Valve—Closure, and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.16	------------------NOTES------------------ 1. Function 2.e digital electronics are excluded. 2. For Functions 3 and 4, the sensor response time may be assumed to be the design sensor response time. 3. For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. 4. For Function 9, the RPS RESPONSE TIME is measured from start of turbine control valve fast closure. --- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – Upscale	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux – Upscale, Setdown	2	3 per logic channel	G	SR 3.3.1.1.2 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Flow Biased Simulated Thermal Power – Upscale	1	3 per logic channel	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ .58W + 62% RTP and ≤ 115.5% RTP(b)
c. Fixed Neutron Flux – Upscale	1	3 per logic channel	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 120% RTP
d. Inop	1,2	3 per logic channel	G	SR 3.3.1.1.7 SR 3.3.1.1.10	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.16	NA

(continued)

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

(b) Allowable Value is .58(W - 5%) + 62% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.16	≤ 1072 psig
4. Reactor Vessel Water Level – Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.16	≥ 157.8 inches
5. Main Steam Isolation Valve – Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.16	≤ 12% closed
6. Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 1.88 psig
7. Scram Discharge Volume Water Level – High					
a. Transmitter/Trip Unit	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 49.5 inches
	5[(a)]	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 49.5 inches
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 49.5 inches
	5[(a)]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 49.5 inches
8. Turbine Stop Valve – Closure	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.16	≤ 7% closed

(continued)

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

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Turbine Control Valve Fast Closure, Trip Oil Pressure – Low	≥ 30% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.16	≥ 465 psig
10. Reactor Mode Switch – Shutdown Position	1,2	2	G	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
	5(a)	2	H	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	4	G	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5(a)	4	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA

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

3.3 INSTRUMENTATION

3.3.1.2 Source Range Monitor (SRM) Instrumentation

LCO 3.3.1.2 The SRM instrumentation in Table 3.3.1.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.2-1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below.	A.1	Restore required SRMs to OPERABLE status.	4 hours
B. Three required SRMs inoperable in MODE 2 with IRMs on Range 2 or below.	B.1	Suspend control rod withdrawal.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	12 hours
D. One or more required SRMs inoperable in MODE 3 or 4.	D.1	Fully insert all insertable control rods.	1 hour
	<u>AND</u>		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Place reactor mode switch in the shutdown position.	1 hour
E. One or more required SRMs inoperable in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion.	Immediately
	<u>AND</u>		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
Refer to Table 3.3.1.2-1 to determine which SRs apply for each applicable MODE or other specified condition.

SURVEILLANCE		FREQUENCY
SR 3.3.1.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2.2	------------------NOTES------------------ 1. Only required to be met during CORE ALTERATIONS. 2. One SRM may be used to satisfy more than one of the following. --- Verify an OPERABLE SRM detector is located in: a. The fueled region; b. The core quadrant where CORE ALTERATIONS are being performed when the associated SRM is included in the fueled region; and c. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region.	12 hours
SR 3.3.1.2.3	Perform CHANNEL CHECK.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2.4	-------------------NOTE------------------- Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. -- Verify count rate is: a. ≥ 3.0 cps with a signal to noise ratio ≥ 2:1; or b. > 1.3 cps with a signal to noise ratio ≥ 5:1.	12 hours during CORE ALTERATIONS AND 24 hours
SR 3.3.1.2.5	-------------------NOTE------------------- The determination of signal to noise ratio is not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. -- Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	7 days
SR 3.3.1.2.6	-------------------NOTE------------------- Not required to be performed until 12 hours after IRMs on Range 2 or below. -- Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2.7	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. Not required to be performed until 12 hours after IRMs on Range 2 or below. --- Perform CHANNEL CALIBRATION.	24 months

Table 3.3.1.2-1 (page 1 of 1)
Source Range Monitor Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Source Range Monitor	2[(a)]	3	SR 3.3.1.2.1 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7
	3,4	2	SR 3.3.1.2.3 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7
	5	2[(b),(c)]	SR 3.3.1.2.1 SR 3.3.1.2.2 SR 3.3.1.2.4 SR 3.3.1.2.5 SR 3.3.1.2.7

(a) With IRMs on Range 2 or below.

(b) Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fueled region includes only that SRM detector.

(c) Special movable detectors may be used in place of SRMs if connected to normal SRM circuits.

3.3 INSTRUMENTATION

3.3.2.1 Control Rod Block Instrumentation

LCO 3.3.2.1 The control rod block instrumentation for each Function in Table 3.3.2.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2.1-1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One rod block monitor (RBM) channel inoperable.	A.1	Restore RBM channel to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two RBM channels inoperable.	B.1	Place one RBM channel in trip.	1 hour
C. Rod worth minimizer (RWM) inoperable during reactor startup.	C.1 <u>OR</u>	Suspend control rod movement except by scram.	Immediately (continued)

ACTIONS

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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more Reactor Mode Switch—Shutdown Position channels inoperable.	E.1	Suspend control rod withdrawal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. 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.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	------------------NOTE------------------ Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	-------------------NOTE------------------- Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.2.1.4	-------------------NOTE------------------- Neutron detectors are excluded. -- Verify the RBM is not bypassed when THERMAL POWER is ≥ 30% RTP and a peripheral control rod is not selected.	24 months
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	24 months
SR 3.3.2.1.6	-------------------NOTE------------------- Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.7	-----------------NOTE----------------- Neutron detectors are excluded. -------------------------------------- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

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

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

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

(b) With THERMAL POWER ≤ 10% RTP.

(c) Reactor mode switch in the shutdown position.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Three channels of feedwater system and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater system and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater system and main turbine high water level trip channels inoperable.	B.1 Restore feedwater system and main turbine high water level trip capability.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1	---------NOTE--------- Only applicable if inoperable channel is the result of an inoperable feedwater pump breaker. --------------------- Remove affected feedwater pump(s) from service.	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided feedwater system and main turbine high water level trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.2.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 203.8 inches.	24 months
SR 3.3.2.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker and valve actuation.	24 months

3.3 INSTRUMENTATION

3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3.1 The PAM instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

1. LCO 3.0.4 is not applicable.
2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1	Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1	Initiate action in accordance with Specification 5.6.6.	Immediately
C. One or more Functions with two required channels inoperable.	C.1	Restore one required channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.3.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	E.1 Be in MODE 3.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	F.1 Initiate action in accordance with Specification 5.6.6.	Immediately

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.3.1.2	Perform CHANNEL CALIBRATION for Functions 9 and 10.	92 days
SR 3.3.3.1.3	Perform CHANNEL CALIBRATION for Functions other than Functions 9 and 10.	24 months

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

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Reactor Vessel Pressure	2	E
2. Reactor Vessel Water Level		
a. Fuel Zone Range	2	E
b. Wide Range	2	E
3. Suppression Pool Water Level		
a. Narrow Range	2	E
b. Wide Range	2	E
4. Drywell Pressure		
a. Narrow Range	2	E
b. Wide Range	2	E
5. Drywell Radiation (High Range)	2	F
6. Drywell Air Temperature	2	E
7. Suppression Chamber Pressure	2	E
8. PCIV Position	2 per penetration flow path(a)(b)	E
9. Drywell H_2 Concentration Analyzer	2	E
10. Drywell O_2 Concentration Analyzer	2	E
11. Suppression Pool Water Temperature	2(c)	E

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

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

(c) Monitoring each suppression pool quadrant.

3.3 INSTRUMENTATION

3.3.3.2 Remote Shutdown System

LCO 3.3.3.2 The Remote Shutdown System Functions shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable.	A.1 Restore required Function to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
When an instrumentation 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.

SURVEILLANCE		FREQUENCY
SR 3.3.3.2.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.3.2.2	Verify each required control circuit and transfer switch is capable of performing the intended functions.	24 months
SR 3.3.3.2.3	Perform CHANNEL CALIBRATION for each required instrumentation channel.	24 months

3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV)—Closure; and

2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure—Low.

<u>OR</u>

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ 30% RTP with any recirculation pump in fast speed.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u>	72 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	--------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- Place channel in trip.	 72 hours
B. One or more Functions with EOC-RPT trip capability not maintained. AND MCPR limit for inoperable EOC-RPT not made applicable.	B.1 OR B.2	Restore EOC-RPT trip capability. Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.	2 hours 2 hours
C. Required Action and associated Completion Time not met.	C.1 OR C.2	Remove the associated recirculation pump fast speed breaker from service. Reduce THERMAL POWER to < 30% RTP.	4 hours 4 hours

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.1.2	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. TSV—Closure: ≤ 7% closed; and b. TCV Fast Closure, Trip Oil Pressure—Low: ≥ 465 psig.	24 months
SR 3.3.4.1.3	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	24 months
SR 3.3.4.1.4	Verify TSV—Closure and TCV Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	24 months
SR 3.3.4.1.5	------------------NOTE------------------ Breaker arc suppression time may be assumed from the most recent performance of SR 3.3.4.1.6. --- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.6	Determine RPT breaker arc suppression time.	60 months

3.3 INSTRUMENTATION

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

LCO 3.3.4.2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

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

b. Reactor Vessel Steam Dome Pressure—High.

APPLICABILITY: MODE 1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status.	14 days
	<u>OR</u>	
	A.2 --------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- Place channel in trip.	14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One Function with ATWS-RPT trip capability not maintained.	B.1	Restore ATWS-RPT trip capability.	72 hours
C. Both Functions with ATWS-RPT trip capability not maintained.	C.1	Restore ATWS-RPT trip capability for one Function.	1 hour
D. Required Action and associated Completion Time not met.	D.1	Remove the associated recirculation pump breaker(s) from service.	6 hours
	<u>OR</u>		
	D.2	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL CHECK.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Calibrate the analog trip modules.	92 days
SR 3.3.4.2.4	Verify, for the Reactor Vessel Steam Dome Pressure—High Function, the low frequency motor generator trip is not bypassed for > 29 seconds when THERMAL POWER is > 5% RTP.	24 months
SR 3.3.4.2.5	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level—Low Low, Level 2: ≥ 101.8 inches; and b. Reactor Vessel Steam Dome Pressure—High: ≤ 1080 psig.	24 months
SR 3.3.4.2.6	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	24 months

3.3 INSTRUMENTATION

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 1.c, 1.d, 2.a, 2.b, 2.c, and 2.d. ---------------------	
		Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	AND		
	B.2	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 3.a and 3.b. ---------------------	
		Declare High Pressure Core Spray (HPCS) System inoperable.	1 hour from discovery of loss of HPCS initiation capability
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3.1	Place channel in trip.	12 hours for Functions 1.a, 1.d, 2.a, and 2.d AND 24 hours for Functions other than Functions 1.a, 1.d, 2.a, and 2.d
	OR		
	B.3.2	--------NOTE--------- Only applicable for Functions 1.a, 1.d, 2.a, and 2.d. --------------------- Isolate the affected flow path(s).	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.e, 1.f, 1.g, 1.h, 1.i, 1.j, 2.e, 2.f, 2.g, 2.h, and 2.i. --------------------- Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>		
	C.2	Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1	--------NOTE--------- Only applicable if HPCS pump suction is not aligned to the suppression pool. --------------------- Declare HPCS System inoperable.	1 hour from discovery of loss of HPCS initiation capability
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2.1	Place channel in trip.	24 hours
	<u>OR</u>		
	D.2.2	Align the HPCS pump suction to the suppression pool.	24 hours
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.k, 1.l, and 2.j. ---------------------	
		Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>		
	E.2	Restore channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	F.1	Declare Automatic Depressurization System (ADS) valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	<u>AND</u>		
	F.2	Place channel in trip.	96 hours from discovery of inoperable channel concurrent with HPCS or reactor core isolation cooling (RCIC) inoperable <u>AND</u> 8 days
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1	--------NOTE--------- Only applicable for Functions 4.b, 4.d, 4.e, 5.b, and 5.d. --------------------- Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. (continued)	G.2 Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCS or RCIC inoperable <u>AND</u> 8 days
H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1 Declare associated supported feature(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.5.1-1 to determine which SRs apply for each ECCS 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 as follows: (a) for up to 6 hours for Functions 3.e, 3.g, 3.h, and 3.i; and (b) for up to 6 hours for Functions other than 3.e, 3.g, 3.h, and 3.i, provided the associated Function or the redundant Function maintains ECCS initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.5.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.5.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.5.1.3	Calibrate the trip unit.	92 days
SR 3.3.5.1.4	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.5.1.5	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.5.1.6	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.5.1-1 (page 1 of 5)
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
1. Low Pressure Coolant Injection-A (LPCI) and Low Pressure Core Spray (LPCS) Subsystems					
a. Reactor Vessel Water Level — Low, Level 3	1,2,3, $4^{(a)}$,$5^{(a)}$	2	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 157.8 inches
b. Reactor Vessel Water Level — Low Low Low, Level 1	1,2,3, $4^{(a)}$,$5^{(a)}$	$2^{(b)}$	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 10.8 inches
c. Drywell Pressure — High	1,2,3	$2^{(b)}$	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 1.88 psig
d. Drywell Pressure — High (Boundary Isolation)	1,2,3	2	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 1.88 psig
e. LPCS Pump Start — Time Delay Relay (Normal Power)	1,2,3 $4^{(a)}$,$5^{(a)}$	1	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 12 seconds
f. LPCI Pump A Start — Time Delay Relay (Normal Power)	1,2,3 $4^{(a)}$,$5^{(a)}$	1	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 7 seconds
g. LPCS Pump Start — Time Delay Relay (Emergency Power)	1,2,3 $4^{(a)}$,$5^{(a)}$	1	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 6.75 seconds
h. LPCI Pump A Start — Time Delay Relay (Emergency Power)	1,2,3, $4^{(a)}$,$5^{(a)}$	1	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2 seconds
i. LPCS Differential Pressure — Low (Injection Permissive)	1,2,3	1	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 40 psid and ≤ 98 psid
	$4^{(a)}$,$5^{(a)}$	1	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 40 psid and ≤ 98 psid

(continued)

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, "ECCS — Shutdown."

(b) Also required to initiate the associated diesel generator (DG).

Table 3.3.5.1-1 (page 2 of 5)
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
1. LPCI A and LPCS Subsystems (continued)					
j. LPCI A Differential Pressure – Low (Injection Permissive)	1,2,3	1	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 70 psid and ≤ 150 psid
	$4^{(a)}, 5^{(a)}$	1	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 70 psid and ≤ 150 psid
k. LPCS Pump Discharge Flow – Low (Bypass)	1,2,3, $4^{(a)}, 5^{(a)}$	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 1000 gpm and ≤ 1440 gpm
l. LPCI Pump A Discharge Flow – Low (Bypass)	1,2,3, $4^{(a)}, 5^{(a)}$	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 770 gpm and ≤ 930 gpm
m. Manual Initiation	1,2,3, $4^{(a)}, 5^{(a)}$	2	C	SR 3.3.5.1.6	NA
2. LPCI B and LPCI C Subsystems					
a. Reactor Vessel Water Level – Low, Level 3	1,2,3, $4^{(a)}, 5^{(a)}$	2	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 157.8 inches
b. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3, $4^{(a)}, 5^{(a)}$	$2^{(b)}$	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 10.8 inches
c. Drywell Pressure – High	1,2,3	$2^{(b)}$	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 1.88 psig

(continued)

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

(b) Also required to initiate the associated DG.

Table 3.3.5.1-1 (page 3 of 5)
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 B and LPCI C Subsystems (continued)					
d. Drywell Pressure – High (Boundary Isolation)	1,2,3	2	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 1.88 psig
e. LPCI Pump B Start – Time Delay Relay (Normal Power)	1,2,3, $4^{(a)}, 5^{(a)}$	1	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 7 seconds
f. LPCI Pump C Start – Time Delay Relay (Normal Power)	1,2,3, $4^{(a)}, 5^{(a)}$	1	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 12 seconds
g. LPCI Pump B Start – Time Delay Relay (Emergency Power)	1,2,3, $4^{(a)}, 5^{(a)}$	1	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2 second
h. LPCI Pump C Start – Time Delay Relay (Emergency Power)	1,2,3 $4^{(a)}, 5^{(a)}$	1	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 7 second
i. LPCI B and C Differential Pressure – Low (Injection Permissive)	1,2,3	1 per valve	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 70 psid and ≤ 150 psid
	$4^{(a)}, 5^{(a)}$	1 per valve	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 70 psid and ≤ 150 psid
j. LPCI Pump B and LPCI Pump C Discharge Flow – Low (Bypass)	1,2,3, $4^{(a)}, 5^{(a)}$	1 per pump	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 770 gpm and ≤ 930 gpm
k. Manual Initiation	1,2,3, $4^{(a)}, 5^{(a)}$	2	C	SR 3.3.5.1.6	NA

(continued)

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

Table 3.3.5.1-1 (page 4 of 5)
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 Core Spray (HPCS) System					
a. Reactor Vessel Water Level — Low Low, Level 2	1,2,3, 4(a),5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 101.8 inches
b. Drywell Pressure — High	1,2,3	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 1.88 psig
c. Reactor Vessel Water Level — High, Level 8	1,2,3, 4(a),5(a)	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 209.3 inches
d. Pump Suction Pressure — Low	1,2,3, 4(c),5(c)	2	D	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 94.5 inches H_2O
e. Pump Suction Pressure — Timer	1,2,3, 4(c),5(c)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 5.5 seconds
f. Suppression Pool Water Level — High	1,2,3	2	D	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 200.7 ft
g. HPCS Pump Discharge Pressure — High (Bypass)	1,2,3, 4(a),5(a)	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 220 psig
h. HPCS System Flow Rate — Low (Bypass)	1,2,3, 4(a),5(a)	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 580 gpm and ≤ 720 gpm
i. Manual Initiation	1,2,3, 4(a),5(a)	2	C	SR 3.3.5.1.6	NA

(continued)

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

(b) Also required to initiate the associated DG.

(c) When HPCS is OPERABLE for compliance with LCO 3.5.2 and aligned to the condensate storage tank while tank water level is not within the limit of SR 3.5.2.2.

Table 3.3.5.1-1 (page 5 of 5)
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
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2(d),3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 10.8 inches
b. ADS Initiation Timer	1,2(d),3(d)	1	G	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≤ 117 seconds
c. Reactor Vessel Water Level – Low, Level 3 (Permissive)	1,2(d),3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 157.8 inches
d. LPCS Pump Discharge Pressure – High	1,2(d),3(d)	2	G	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 125 psig and ≤ 150 psig
e. LPCI Pump A Discharge Pressure – High	1,2(d),3(d)	2	G	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 115 psig and ≤ 130 psig
f. Manual Initiation	1,2(d),3(d)	4	G	SR 3.3.5.1.6	NA
5. ADS Trip System B					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2(d),3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 10.8 inches
b. ADS Initiation Timer	1,2(d),3(d)	1	G	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≤ 117 seconds
c. Reactor Vessel Water Level – Low, Level 3 (Permissive)	1,2(d),3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 157.8 inches
d. LPCI Pumps B & C Discharge Pressure – High	1,2(d),3(d)	2 per pump	G	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 115 psig and ≤ 130 psig
e. Manual Initiation	1,2(d),3(d)	4	G	SR 3.3.5.1.6	NA

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

3.3 INSTRUMENTATION

3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation

LCO 3.3.5.2
The RCIC System instrumentation for each Function in Table 3.3.5.2-1 shall be OPERABLE.

APPLICABILITY:
MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

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

1. Separate Condition entry is allowed for each channel.

2. When the Function 2 channels are placed in an inoperable status solely for performance of SR 3.5.3.4, entry into associated Conditions and Required Actions is not required.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.2-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	B.1 Declare RCIC System inoperable. AND B.2 Place channel in trip.	1 hour from discovery of loss of RCIC initiation capability 24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	C.1 Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	D.1 --------NOTE--------- Only applicable if RCIC pump suction is not aligned to the suppression pool. --------------------- Declare RCIC System inoperable.	1 hour from discovery of loss of RCIC initiation capability
	<u>AND</u>	
	D.2.1 Place channel in trip.	24 hours
	<u>OR</u>	
	D.2.2 Align RCIC pump suction to the suppression pool.	24 hours
E. Required Action and associated Completion Time of Condition B, C, or D not met.	E.1 Declare RCIC System inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.5.2-1 to determine which SRs apply for each RCIC 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 as follows: (a) for up to 6 hours for Functions 4 and 5; and (b) for up to 6 hours for Functions 1, 2, and 3 provided the associated Function maintains RCIC initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.5.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.5.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.5.2.3	Calibrate the trip units.	92 days
SR 3.3.5.2.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.5.2.5	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

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 3.3.5.2.4 SR 3.3.5.2.5	≥ 101.8 inches
2. Reactor Vessel Water Level – High, Level 8	4	B	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4 SR 3.3.5.2.5	≤ 209.3 inches
3. Pump Suction Pressure – 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 3.3.5.2.5	≥ 101 inches Wg
4. Pump Suction Pressure – Timer	1	D	SR 3.3.5.2.2 SR 3.3.5.2.4 SR 3.3.5.2.5	≤ 12.3 seconds
5. Manual Initiation	2	C	SR 3.3.5.2.5	NA

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.b, 5.b, and 5.c AND 24 hours for Functions other than Functions 2.b, 5.b, and 5.c
B. One or more automatic Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1	Isolate associated main steam line (MSL).	12 hours
	<u>OR</u>		
	D.2.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	D.2.2	Be in MODE 4.	36 hours
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1	Be in MODE 2.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1	Isolate the affected penetration flow path(s).	1 hour
G. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	G.1	Isolate the affected penetration flow path(s).	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Required Action and associated Completion Time of Condition F or G not met. OR As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	H.1 Be in MODE 3. AND H.2 Be in MODE 4.	12 hours 36 hours
I. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	I.1 Declare associated standby liquid control (SLC) subsystem inoperable. OR I.2 Isolate the Reactor Water Cleanup (RWCU) System.	1 hour 1 hour
J. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	J.1 Initiate action to restore channel to OPERABLE status. OR J.2 Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling (SDC) System.	Immediately Immediately

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	------------------NOTE------------------ Only required to be met when the Allowable Value is adjusted in accordance with Table 3.3.6.1-1, footnote(b). -- Verify the actual ambient temperature reading for all OPERABLE channels is $\geq T_{amb}$.	12 hours
SR 3.3.6.1.3	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.1.4	Calibrate the trip unit.	92 days
SR 3.3.6.1.5	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.1.6	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.7	-------------------NOTE------------------- The sensor response time may be assumed to be the design sensor response time. -- Verify the ISOLATION SYSTEM RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level — Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 10.8 inches
b. Main Steam Line Pressure — Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 746 psig
c. Main Steam Line Flow — High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 122.8 psid
d. Condenser Vacuum — Low	1,2[(a)], 3[(a)]	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 7.6 inches Hg vacuum
e. Main Steam Line Tunnel Temperature — High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 170.6°F
f. Main Steam Line Tunnel Differential Temperature — High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 71.7°F
g. Main Steam Line Tunnel Lead Enclosure Temperature — High	1,2,3	2 per area	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 151.6°F[(b)]
h. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA

(continued)

(a) With any turbine stop valve not closed.

(b) 151.6°F + (0.6)(T_{amb} - 90°F) and ≤ 175.6°F provided the absence of steam leaks in the main steam line tunnel lead enclosure area is verified by visual inspection prior to establishing the Allowable Value.

Table 3.3.6.1-1 (page 2 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment Isolation					
a. Reactor Vessel Water Level – Low Low, Level 2	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 101.8 inches
b. Drywell Pressure – High	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psig
c. Standby Gas Treatment (SGT) System Exhaust Radiation – High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.0×10^{-2} µCi/cc with time delay ≤ 18.5 seconds
d. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA
3. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 175.6 inches water
b. RCIC Steam Line Flow – Timer	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 13 seconds
c. RCIC Steam Supply Pressure – Low	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 70 psia
d. RCIC Turbine Exhaust Diaphragm Pressure – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig
e. RCIC Equipment Room Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140.5°F
f. RCIC Steam Line Tunnel Temperature – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140.5°F

(continued)

Table 3.3.6.1-1 (page 3 of 5)
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. RCIC System Isolation (continued)					
g. RHR Equipment Room Area Temperature – High	1,2,3	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 144.5°F
h. Reactor Building Pipe Chase Area Temperature – High	1,2,3	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	
El. ≈ 319 ft.					≤ 144.5°F
El. ≈ 292 ft.					≤ 140.5°F
El. ≈ 266 ft.					≤ 140.5°F
El. ≈ 227 ft.					≤ 140.5°F
i. Reactor Building General Area Temperature – High	1,2,3	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 134°F
j. Area Temperature – Timer	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.15 seconds
k. RCIC/RHR Steam Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 40.73 inches water
l. RCIC/RHR Steam Flow – Timer	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 13 seconds
m. Manual Initiation	1,2,3	1[c]	G	SR 3.3.6.1.6	NA
4. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 165.5 gpm
b. Differential Flow – Timer	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 47 seconds

(continued)

(c) Only inputs into one of the two trip systems.

Table 3.3.6.1-1 (page 4 of 5)
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. RWCU System Isolation (continued)					
c. Heat Exchanger Room Temperature – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140.5°F
d. Pump Room Temperature – High	1,2,3	1 per room	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	
Pump Room A					≤ 144.5°F
Pump Room B					≤ 159.5°F
e. Reactor Building Pipe Chase Area Temperature – High	1,2,3	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	
El. ≈ 319 ft.					≤ 144.5°F
El. ≈ 292 ft.					≤ 140.5°F
El. ≈ 266 ft.					≤ 140.5°F
El. ≈ 227 ft.					≤ 140.5°F
f. Reactor Vessel Water Level – Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 101.8 inches
g. SLC System Initiation	1,2	1	I	SR 3.3.6.1.6	NA
h. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA
5. RHR SDC System Isolation					
a. RHR Equipment Room Area Temperature – High	3	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 144.5°F

(continued)

Table 3.3.6.1-1 (page 5 of 5)
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. RHR SDC System Isolation (continued)					
b. Reactor Vessel Water Level – Low, Level 3	3,4,5	2[(d)]	J	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 157.8 inches
c. Reactor Vessel Pressure – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 148 psig
d. Reactor Building Pipe Chase Area Temperature – High	3	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	
El. ≈ 319 ft.					≤ 144.5°F
El. ≈ 292 ft.					≤ 140.5°F
El. ≈ 266 ft.					≤ 140.5°F
El. ≈ 227 ft.					≤ 140.5°F
e. Reactor Building General Area Temperature – High	3	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 134°F
f. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA

(d) Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

3.3 INSTRUMENTATION

3.3.6.2 Secondary Containment Isolation Instrumentation

LCO 3.3.6.2 The secondary containment isolation instrumentation for each Function in Table 3.3.6.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.2-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place channel in trip.	12 hours for Function 2 <u>AND</u> 24 hours for Functions other than Function 2
B. One or more Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour
C. Required Action and associated Completion Time not met.	C.1.1 Isolate the associated penetration flow path(s). <u>OR</u>	1 hour (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.1.2 Declare associated secondary containment isolation valves inoperable.	1 hour
	<u>AND</u>	
	C.2.1 Place the associated standby gas treatment (SGT) subsystem in operation.	1 hour
	<u>OR</u>	
	C.2.2 Declare associated SGT subsystem inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.6.2-1 to determine which SRs apply for each Secondary 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 3.3.6.2.1 Perform CHANNEL CHECK.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.2.3	Calibrate the trip unit.	92 days
SR 3.3.6.2.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.2.5	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES AND OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level – Low Low, Level 2	1,2,3,(a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 101.8 inches
2. Drywell Pressure – High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.88 psig
3. Reactor Building Above the Refuel Floor Exhaust Radiation – High	1,2,3, (a),(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.46×10^{-3} µCi/cc
4. Reactor Building Below the Refuel Floor Exhaust Radiation – High	1,2,3, (a),(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.46×10^{-3} µCi/cc

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

(b) During CORE ALTERATIONS, and during movement of irradiated fuel assemblies in the secondary containment.

3.3 INSTRUMENTATION

3.3.7.1 Control Room Envelope Filtration (CREF) System Instrumentation

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

APPLICABILITY: According to Table 3.3.7.1-1.

ACTIONS

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

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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	C.1 Declare associated CREF subsystem inoperable.	1 hour from discovery of loss of CREF initiation capability in both trip systems
	<u>AND</u>	
	C.2 Place channel in trip.	12 hours
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Place the associated CREF subsystem in the emergency pressurization mode of operation.	1 hour
	<u>OR</u>	
	D.2 Declare associated CREF subsystem inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.7.1-1 to determine which SRs apply for each 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 CREF initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Calibrate the trip units.	92 days
SR 3.3.7.1.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.7.1.5	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.7.1-1 (page 1 of 1)
Control Room Envelope Filtration System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level – Low Low, Level 2	1,2,3, (a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≥ 101.8 inches
2. Drywell Pressure – High	1,2,3	2	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 1.88 psig
3. Main Control Room Ventilation Radiation Monitor – High	1,2,3, (a),(b)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 5.92×10^{-6} µCi/cc

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

(b) During CORE ALTERATIONS, and during movement of irradiated fuel assemblies in the secondary containment.

3.3 INSTRUMENTATION

3.3.7.2 Mechanical Vacuum Pump Isolation Instrumentation

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

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

ACTIONS

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

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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1	Isolate the associated mechanical vacuum pump(s).	12 hours
	<u>OR</u>		
	C.2	Remove the associated mechanical vacuum pump breaker(s) from service.	12 hours
	<u>OR</u>		
	C.3	Isolate the main steam lines.	12 hours
	<u>OR</u>		
	C.4	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 3.6 x full power background.	24 months
SR 3.3.7.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including isolation valve and mechanical vacuum pump breakers actuation.	24 months

3.3 INSTRUMENTATION

3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When the associated diesel generator (DG) is required to be OPERABLE by LCO 3.8.2, "AC Sources—Shutdown."

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.8.1-1 to determine which SRs apply for each LOP 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 2 hours provided the associated Function maintains LOP initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.8.1.1	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.8.1.2	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.8.1.3	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

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

FUNCTION	REQUIRED CHANNELS PER DIVISION	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Divisions 1 and 2 – 4.16 kV Emergency Bus Undervoltage			
a. Loss of Voltage – 4.16 kV Basis	2	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 2950 V and ≤ 3468 V
b. Loss of Voltage – Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 2.80 seconds and ≤ 3.20 seconds
c. Degraded Voltage – 4.16 kV Basis	2	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 3820 V and ≤ 3898 V
d. Degraded Voltage – Time Delay, No LOCA	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 27.8 seconds and ≤ 32.2 seconds
e. Degraded Voltage – Time Delay, LOCA	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 7.4 seconds and ≤ 8.6 seconds
2. Division 3 – 4.16 kV Emergency Bus Undervoltage			
a. Loss of Voltage – 4.16 kV Basis	2	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 2950 V and ≤ 3468 V
b. Loss of Voltage – Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 2.8 seconds and ≤ 3.2 seconds
c. Degraded Voltage – 4.16 kV Basis	2	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 3820 V and ≤ 3898 V
d. Degraded Voltage – Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 11.0 seconds and ≤ 13.0 seconds

3.3 INSTRUMENTATION

3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring—Logic

LCO 3.3.8.2 Two RPS electric power monitoring assemblies shall be OPERABLE for each RPS logic bus.

APPLICABILITY: MODES 1, 2, and 3,
MODES 4 and 5 with both residual heat removal (RHR) shutdown cooling (SDC) suction isolation valves open,
MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both RPS logic buses with one electric power monitoring assembly inoperable.	A.1 Restore electric power monitoring assembly(s) to OPERABLE status.	72 hours
B. One or both RPS logic buses with both electric power monitoring assemblies inoperable.	B.1 Restore electric power monitoring assemblies to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met in MODE 4 or 5 with both RHR SDC suction isolation valves open.	D.1	Initiate action to restore one electric power monitoring assembly to OPERABLE status for each RPS logic bus.	Immediately
	<u>OR</u>		
	D.2	Initiate action to isolate the RHR SDC System.	Immediately
E. Required Action and associated Completion Time of Condition A or B not met in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.	E.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
F. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	F.1.1	Isolate the associated secondary containment penetration flow path(s).	Immediately
	<u>OR</u>		
	F.1.2	Declare associated secondary containment isolation valves inoperable.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. (continued)	F.2.1 Place the associated standby gas treatment (SGT) subsystem(s) in operation.	Immediately
	<u>OR</u>	
	F.2.2 Declare associated SGT subsystem(s) inoperable.	Immediately
	<u>AND</u>	
	F.3.1 Place the associated control room envelope filtration (CREF) subsystem(s) in the emergency pressurization mode of operation.	Immediately
	<u>OR</u>	
	F.3.2 Declare associated CREF subsystem(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

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

When an RPS electric power monitoring assembly is placed in an inoperable status solely for performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 6 hours provided the other RPS electric power monitoring assembly for the associated RPS logic bus maintains trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.8.2.1	Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.8.2.2	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Overvoltage (with time delay set to ≤ 2.5 seconds) Bus A ≤ 133.8 V Bus B ≤ 133.8 V b. Undervoltage (with time delay set to ≤ 2.5 seconds) Bus A ≥ 115.5 V Bus B ≥ 114.2 V c. Underfrequency (with time delay set to ≤ 2.5 seconds) Bus A ≥ 57.5 Hz Bus B ≥ 57.5 Hz	24 months
SR 3.3.8.2.3	Perform a system functional test.	24 months

3.3 INSTRUMENTATION

3.3.8.3 Reactor Protection System (RPS) Electric Power Monitoring—Scram Solenoids

LCO 3.3.8.3 Two RPS electric power monitoring assemblies shall be OPERABLE for each RPS scram solenoid bus.

APPLICABILITY: MODES 1 and 2,
MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or both RPS scram solenoid buses with one electric power monitoring assembly inoperable.	A.1	Remove associated RPS scram solenoid bus(es) from service.	72 hours
B. One or both RPS scram solenoid buses with both electric power monitoring assemblies inoperable.	B.1	Remove associated RPS scram solenoid bus(es) from service.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1 or 2.	C.1	Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.	D.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

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

When an RPS electric power monitoring assembly is placed in an inoperable status solely for performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 6 hours provided the other RPS electric power monitoring assembly for the associated RPS scram solenoid bus maintains trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.8.3.1	Perform CHANNEL FUNCTIONAL TEST.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.8.3.2	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Overvoltage (with time delay set to ≤ 2.5 seconds) Bus A ≤ 130.5 V Bus B ≤ 131.7 V b. Undervoltage (with time delay set to ≤ 2.5 seconds) Bus A ≥ 113.0 V Bus B ≥ 113.6 V c. Underfrequency (with time delay set to ≤ 2.5 seconds) Bus A ≥ 57.5 Hz Bus B ≥ 57.5 Hz	24 months
SR 3.3.8.3.3	Perform a system functional test.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation in the "Unrestricted Zone" of Figure 3.4.1-1,

<u>OR</u>

One recirculation loop shall be in operation in the "Unrestricted Zone" of Figure 3.4.1-1 with the following limits applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power—Upscale), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation; and

d. LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 1.a (Rod Block Monitor—Upscale), Allowable Value of Table 3.3.2.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. No recirculation loops in operation.	A.1	Restore operation to the "Unrestricted Zone" of Figure 3.4.1-1.	2 hours
	<u>AND</u>		
	A.2	Be in MODE 2.	6 hours
	<u>AND</u>		
	A.3	Be in MODE 3.	12 hours
B. One or two recirculation loops in operation in the "Restricted Zone" of Figure 3.4.1-1 with core flow > 39% of rated core flow.	B.1	Verify APRM and LPRM neutron flux noise levels ≤ 3 times baseline noise levels.	Once per 8 hours <u>AND</u> 30 minutes after completion of any THERMAL POWER increase ≥ 5% RTP
	<u>AND</u>		
	B.2	Determine core flow.	Once per 12 hours
C. Required Action B.1 and associated Completion Time not met.	C.1	Restore APRM and LPRM neutron flux noise levels to ≤ 3 times baseline noise levels.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action B.2 and associated Completion Time not met. OR One or two recirculation loops in operation in the "Restricted Zone" of Figure 3.4.1-1 with core flow ≤ 39% of rated core flow.	D.1	Reduce THERMAL POWER to the "Unrestricted Zone" of Figure 3.4.1-1.	4 hours
	OR		
	D.2	Increase core flow to > 39% of rated core flow.	4 hours
E. Recirculation loop flow mismatch not within limits.	E.1	Declare the recirculation loop with lower flow to be "not in operation."	2 hours
F. Requirements of the LCO not met for reasons other than Conditions A, B, C, D, and E.	F.1	Satisfy the requirements of the LCO.	4 hours
G. Required Action and associated Completion Time of Condition F not met.	G.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify operation is in the "Unrestricted Zone" of Figure 3.4.1-1.	12 hours
SR 3.4.1.2	-------------------NOTE------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. -- Verify jet pump loop flow mismatch with both recirculation loops in operation is: a. $\leq$ 10% of rated core flow when operating at an effective core flow < 70% of rated core flow; and b. $\leq$ 5% of rated core flow when operating at an effective core flow $\geq$ 70% of rated core flow.	24 hours

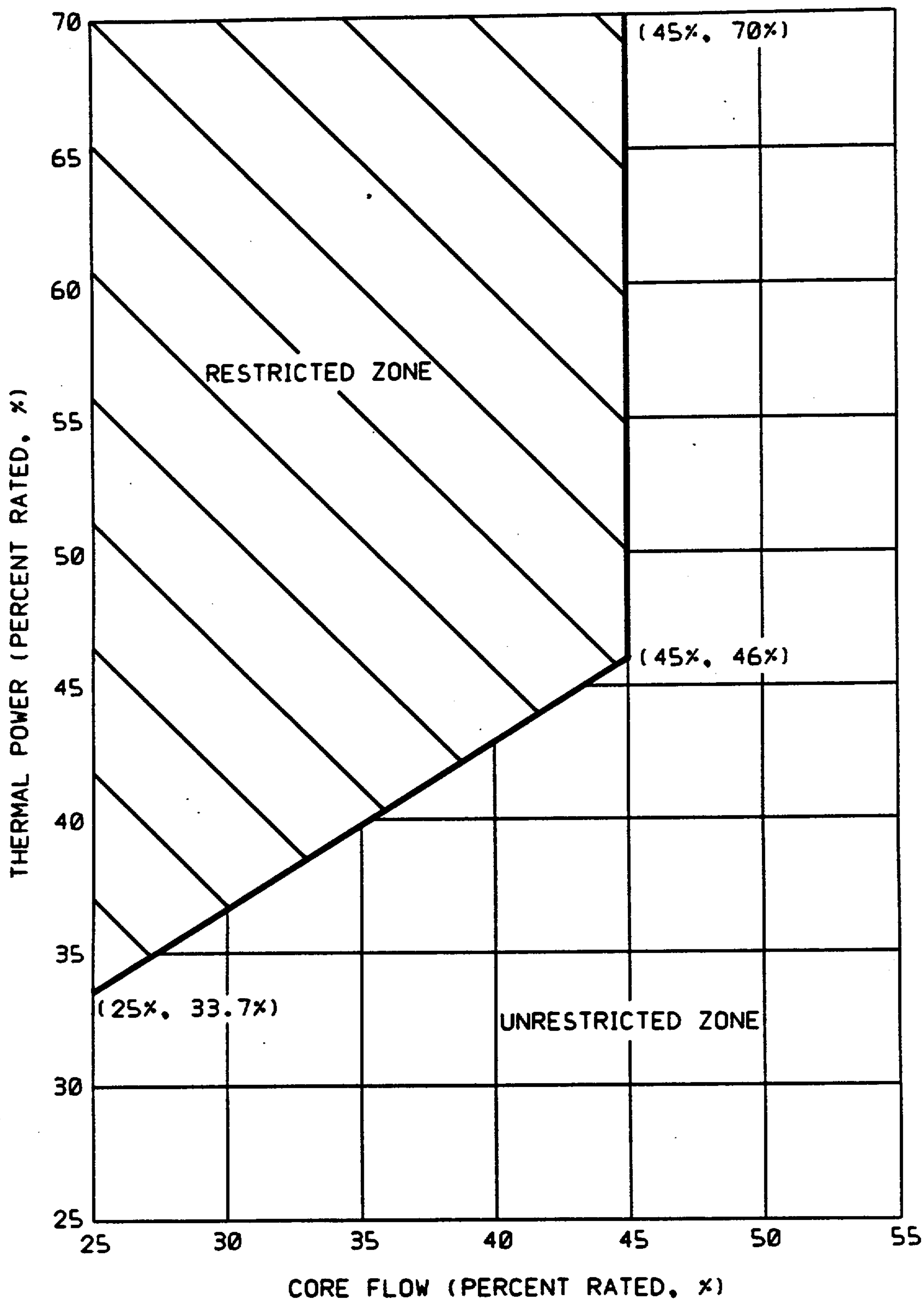

Figure 3.4.1-1 (Page 1 of 1)
Stability Monitoring Power-to-Flow Map

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Flow Control Valves (FCVs)

LCO 3.4.2 A recirculation loop FCV shall be OPERABLE in each operating recirculation loop.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required FCVs inoperable.	A.1 Lock up the FCV.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify each FCV fails "as is" on loss of hydraulic pressure at the hydraulic unit.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.2.2	Verify average rate of each FCV movement is: a. ≤ 11% of stroke per second for opening; and b. ≤ 11% of stroke per second for closing.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Jet Pumps

LCO 3.4.3 All jet pumps shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more jet pumps inoperable.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-------------------NOTES------------------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > 25% RTP. --- Verify at least two of the following criteria (a, b, and c) are satisfied for each operating recirculation loop: a. Jet pump loop flow versus flow control valve position differs by ≤ 10% from established patterns. b. Jet pump loop flow versus recirculation loop drive flow differs by ≤ 10% from established patterns. c. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.	24 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Safety/Relief Valves (S/RVs)

LCO 3.4.4 The safety function of 16 S/RVs shall be OPERABLE,

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required S/RVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify the safety function lift setpoints of the required S/RVs are as follows: Number of S/RVs — Setpoint (psig) 2 — 1165 psig ± 35.0 4 — 1175 psig ± 35.0 4 — 1185 psig ± 36.0 4 — 1195 psig ± 36.0 4 — 1205 psig ± 36.0 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Operational LEAKAGE

LCO 3.4.5 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE;

c. ≤ 25 gpm identified LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Unidentified LEAKAGE not within limit. OR Identified LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce unidentified LEAKAGE increase to within limit. OR B.2 Identify source of unidentified LEAKAGE increase.	4 hours 4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify RCS unidentified and identified LEAKAGE and unidentified LEAKAGE increase are within limits.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.6 The leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1 and 2,
MODE 3, except valves in the residual heat removal shutdown cooling flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

ACTIONS

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

1. Separate Condition entry is allowed for each flow path.
2. Enter applicable Conditions and Required Actions for systems made inoperable by PIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	------------NOTE------------ Each valve used to satisfy Required Action A.1 and Required Action A.2 shall have been verified to meet SR 3.4.6.1 and be in the reactor coolant pressure boundary or the high pressure portion of the system. ----------------------------	(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	<u>AND</u>		
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	-------------------NOTE------------------- Only required to be performed in MODES 1 and 2. -- Verify equivalent leakage of each RCS PIV is ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure ≥ 1000 psig and ≤ 1040 psig.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Leakage Detection Instrumentation

LCO 3.4.7 The following RCS leakage detection instrumentation shall be OPERABLE:

a. Drywell floor drain tank fill rate monitoring system; and

b. One channel of either drywell atmospheric particulate or atmospheric gaseous monitoring system.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain tank fill rate monitoring system inoperable.	-------------NOTE------------ LCO 3.0.4 is not applicable. -----------------------------	
	A.1 Restore drywell floor drain tank fill rate monitoring system to OPERABLE status.	30 days
B. Required drywell atmospheric monitoring system inoperable.	-------------NOTE------------ LCO 3.0.4 is not applicable. -----------------------------	
	B.1 Analyze grab samples of drywell atmosphere.	Once per 12 hours
	<u>AND</u>	
	B.2 Restore required drywell atmospheric monitoring system to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 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 3.4.7.1	Perform CHANNEL CHECK of required drywell atmospheric monitoring system.	12 hours
SR 3.4.7.2	Perform CHANNEL FUNCTIONAL TEST of the drywell floor drain tank fill rate monitoring system.	31 days
SR 3.4.7.3	Perform source check of required drywell atmospheric monitoring system.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.7.4	Perform CHANNEL FUNCTIONAL TEST of required drywell atmospheric monitoring system.	184 days
SR 3.4.7.5	Perform CHANNEL CALIBRATION of required leakage detection instrumentation.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Specific Activity

LCO 3.4.8 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity $\leq 0.2\ \mu Ci/gm$.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor coolant specific activity $> 0.2\ \mu Ci/gm$ and $\leq 4.0\ \mu Ci/gm$ DOSE EQUIVALENT I-131.	-------------NOTE----------- LCO 3.0.4 is not applicable. ---------------------------	
	A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
<u>OR</u>	<u>AND</u>	
Reactor coolant Specific activity $> 4.0\ \mu Ci/gm$ DOSE EQUIVALENT I-131.	B.2.1 Isolate all main steam lines.	12 hours
	<u>OR</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.2.1 Be in MODE 3. AND B.2.2.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	------------------NOTE------------------ Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity is $\leq 0.2\ \mu Ci/gm$.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

LCO 3.4.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

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

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

2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for performance of Surveillances.

APPLICABILITY: MODE 3 with reactor steam dome pressure less than the RHR cut-in permissive pressure.

ACTIONS

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

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1	Initiate action to restore RHR shutdown cooling subsystem to OPERABLE status.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour
	<u>AND</u>	
	A.3 Be in MODE 4.	24 hours
B. No RHR shutdown cooling subsystem in operation. <u>AND</u> No recirculation pump in operation.	B.1 Initiate action to restore one RHR shutdown cooling subsystem or one recirculation pump to operation.	Immediately
	<u>AND</u>	
	B.2 Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>	
	B.3 Monitor reactor coolant temperature and pressure.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-------------------NOTE------------------- Not required to be met until 2 hours after reactor steam dome pressure is less than the RHR cut-in permissive pressure. -- Verify one RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

LCO 3.4.10 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

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

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

2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

APPLICABILITY: MODE 4.

ACTIONS

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

Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

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

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.10.1 Verify one RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.11 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation loop temperature requirements shall be maintained within limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. ---------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	<u>AND</u>		
	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations, and RCS system leakage and hydrostatic testing. --- Verify: a. RCS pressure and RCS temperature are within the applicable limits specified in Figures 3.4.11-1, 3.4.11-2, 3.4.11-3, 3.4.11-4, and 3.4.11-5; b. RCS heatup and cooldown rates are ≤ 100°F in any 1 hour period; and c. RCS temperature change during system leakage and hydrostatic testing is ≤ 20°F in any 1 hour period when the RCS pressure and RCS temperature are not within the limits of Figure 3.4.11-2 or Figure 3.4.11-3, as applicable.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.2	Verify RCS pressure and RCS temperature are within the applicable criticality limits specified in Figures 3.4.11-4 and 3.4.11-5.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.11.3	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 145°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.11.4	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.5	-------------------NOTE------------------- Only required to be met in single loop operation with THERMAL POWER ≤ 30% RTP or the operating jet pump loop flow ≤ 50% rated jet pump loop flow. -- Verify the difference between the bottom head coolant temperature and the RPV coolant temperature is ≤ 145°F.	Once within 15 minutes prior to an increase in THERMAL POWER or an increase in jet pump loop flow
SR 3.4.11.6	-------------------NOTE------------------- Only required to be met in single loop operation when the idle recirculation loop is not isolated from the RPV, and with THERMAL POWER ≤ 30% RTP or the operating jet pump loop flow ≤ 50% rated jet pump loop flow. -- Verify the difference between the reactor coolant temperature in the recirculation loop not in operation and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to an increase in THERMAL POWER or an increase in jet pump loop flow
SR 3.4.11.7	-------------------NOTE------------------- Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify reactor vessel flange and head flange temperatures are ≥ 70°F.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.8	-------------------NOTE------------------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. -- Verify reactor vessel flange and head flange temperatures are ≥ 70°F.	30 minutes
SR 3.4.11.9	-------------------NOTE------------------- Not required to be performed until 12 hours after RCS temperature ≤ 90°F in MODE 4. -- Verify reactor vessel flange and head flange temperatures are ≥ 70°F.	12 hours

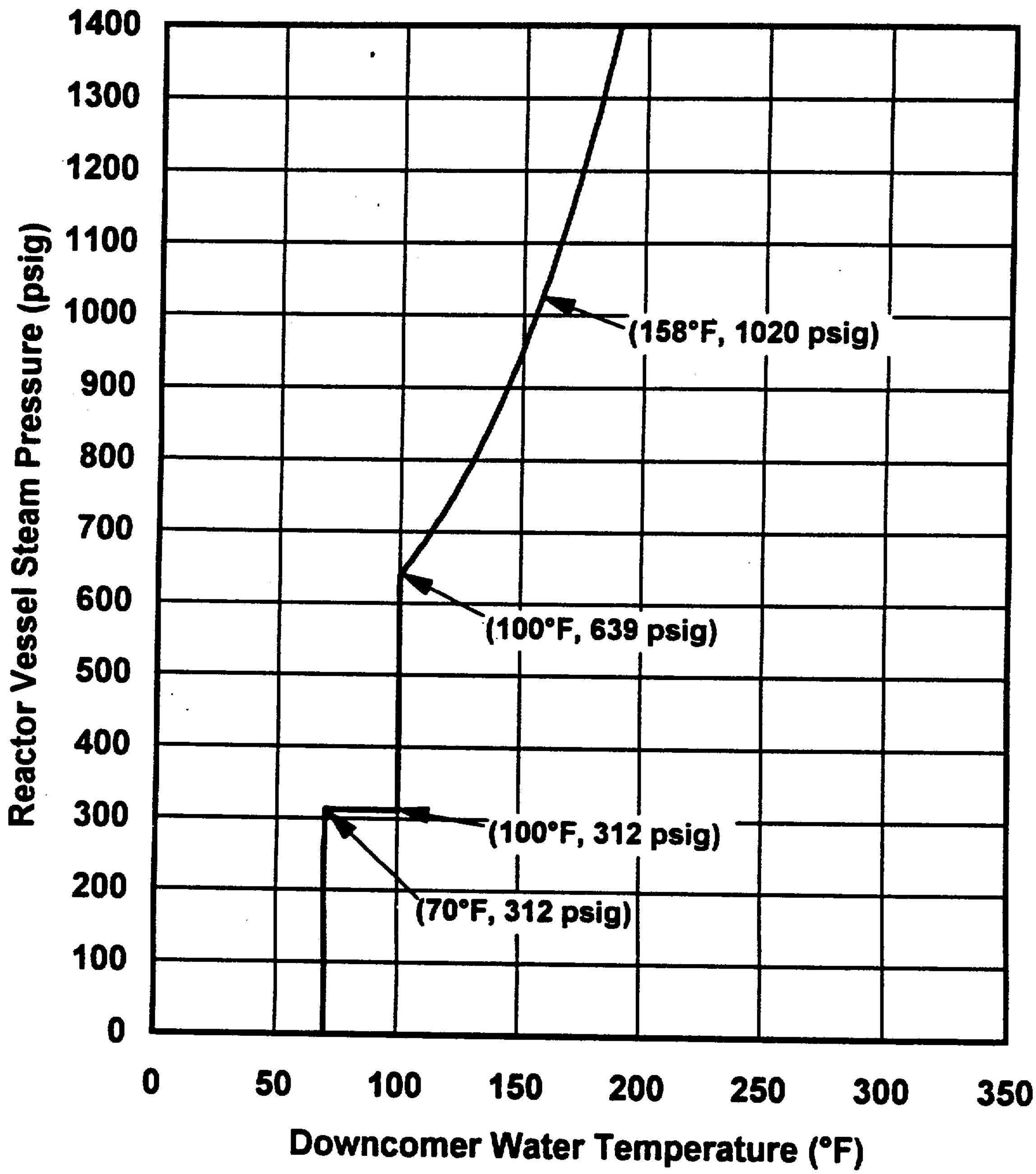

Figure 3.4.11-1 (Page 1 of 1)
Non-Nuclear System Leakage and Hydrostatic Testing Curve

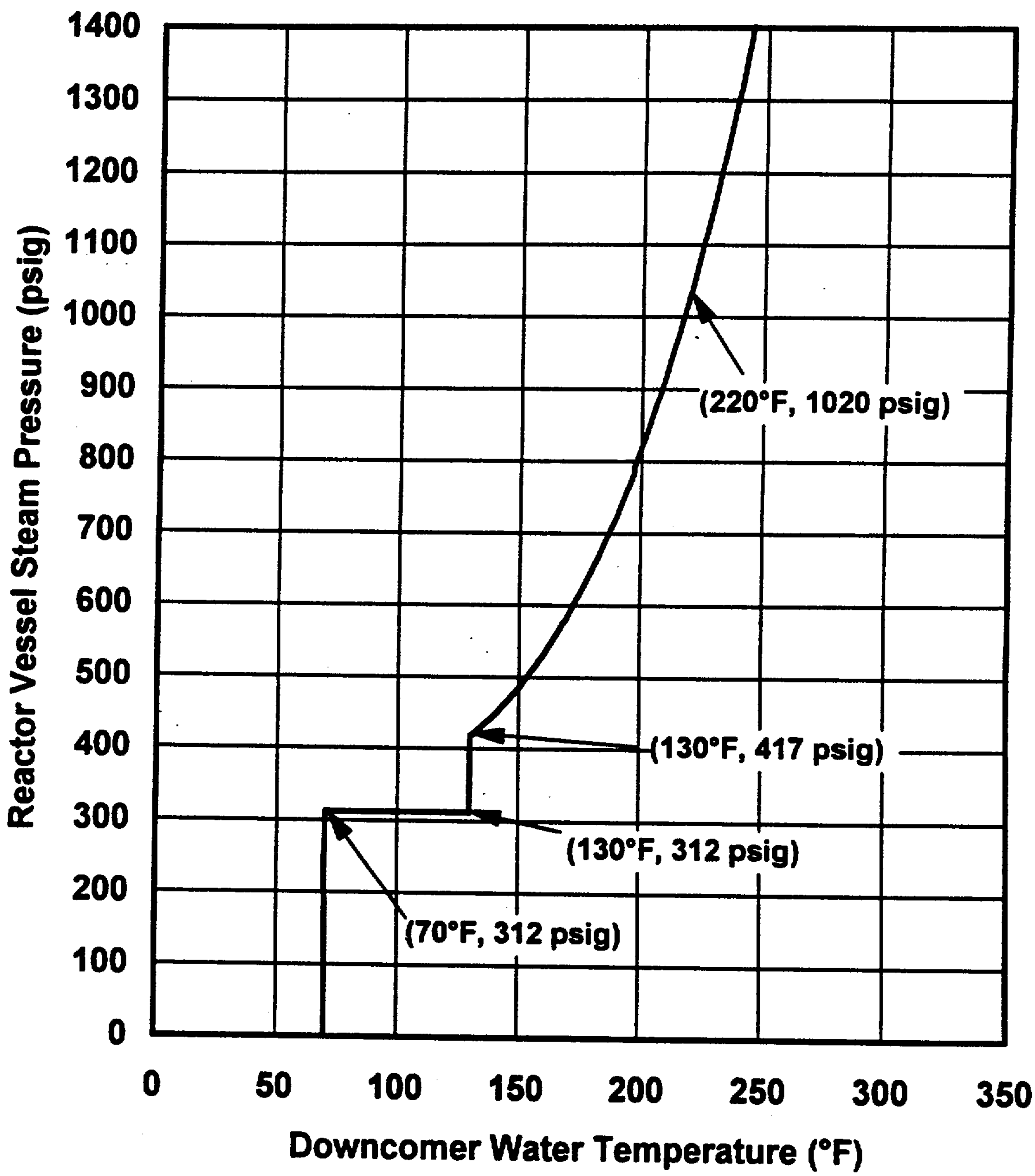

Figure 3.4.11-2 (Page 1 of 1)
Non-Nuclear Heatup Curve

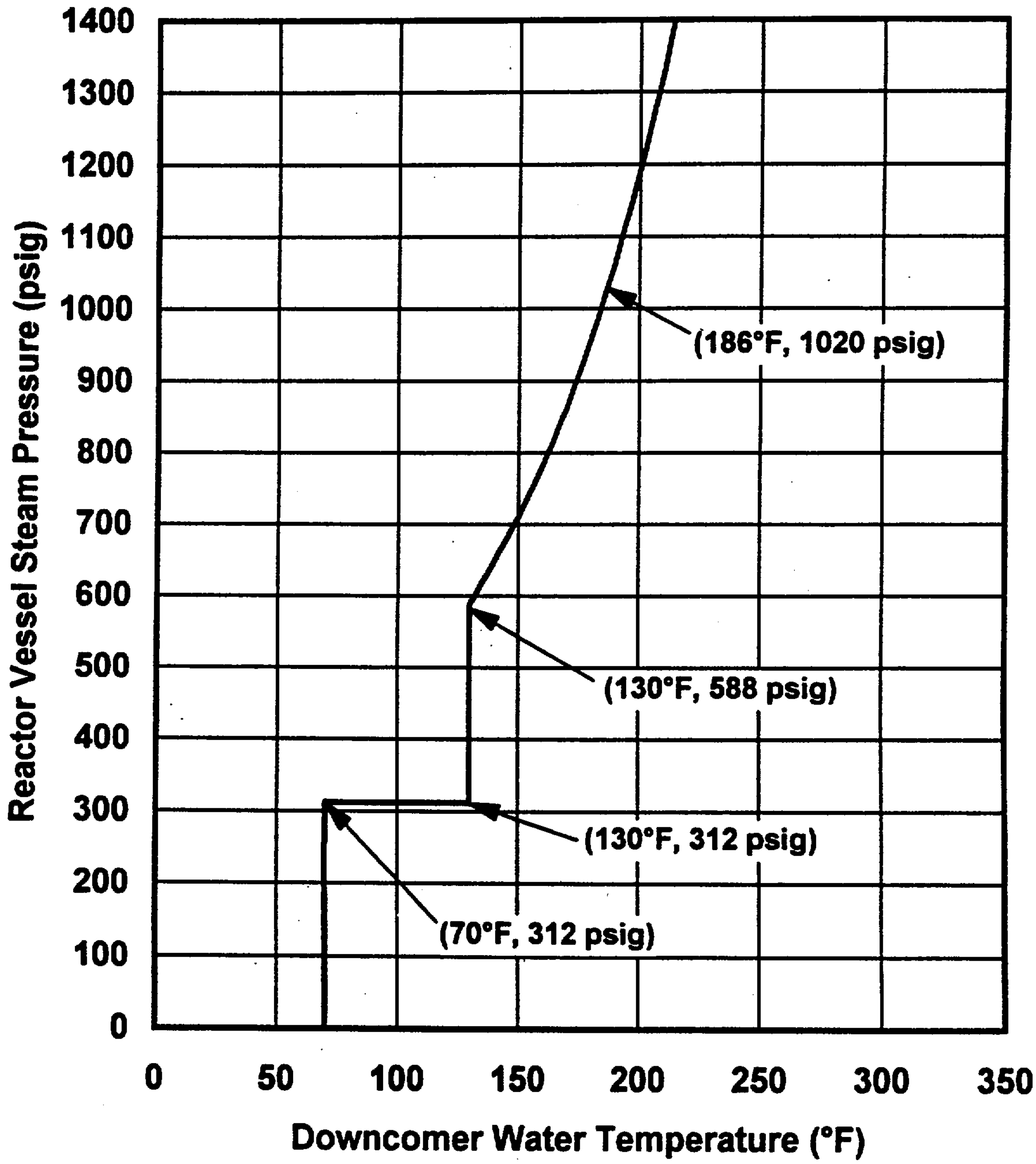

Figure 3.4.11-3 (Page 1 of 1)
Non-Nuclear Cooldown Curve

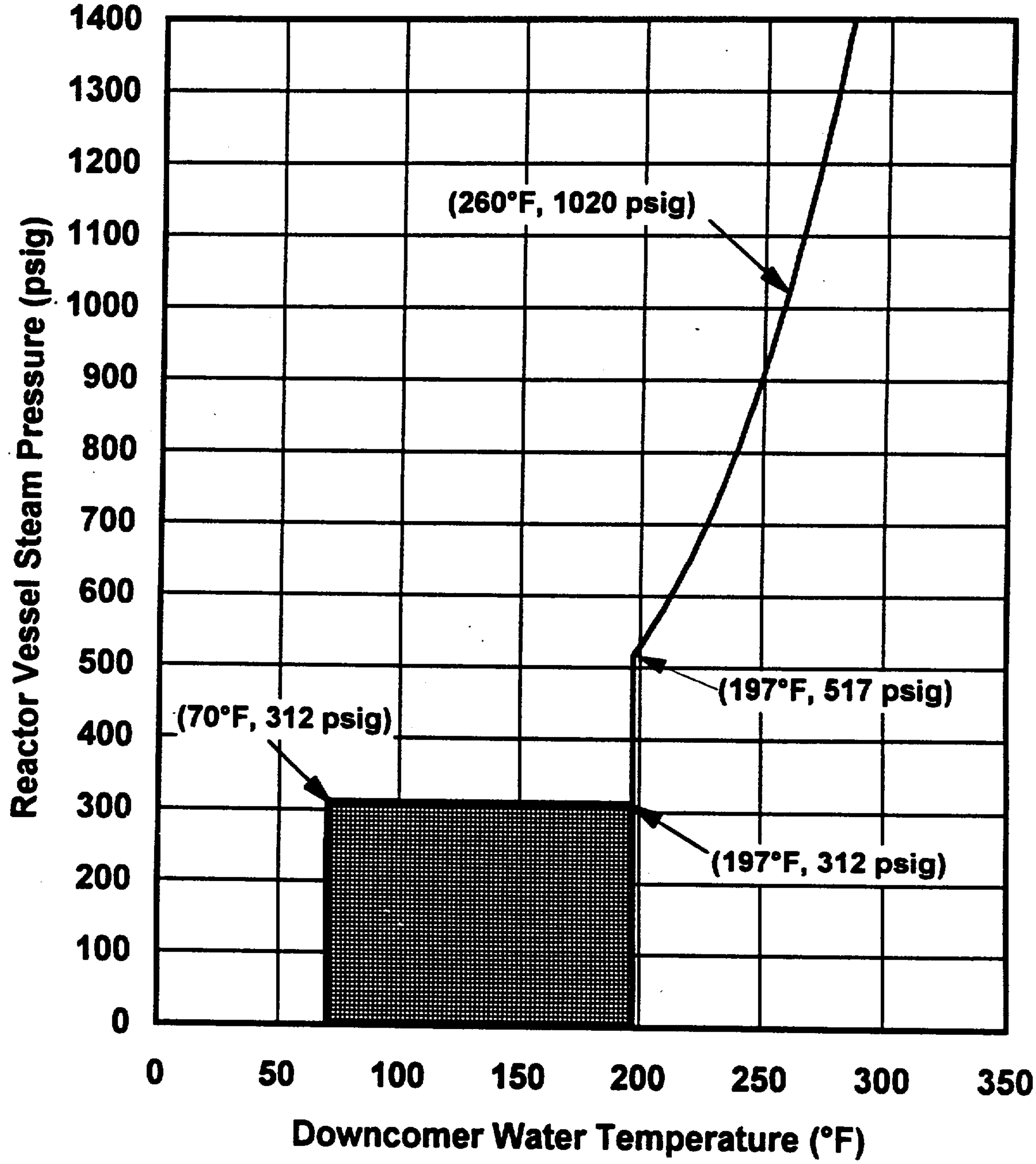

Figure 3.4.11-4 (Page 1 of 1)
Nuclear Heatup Curve

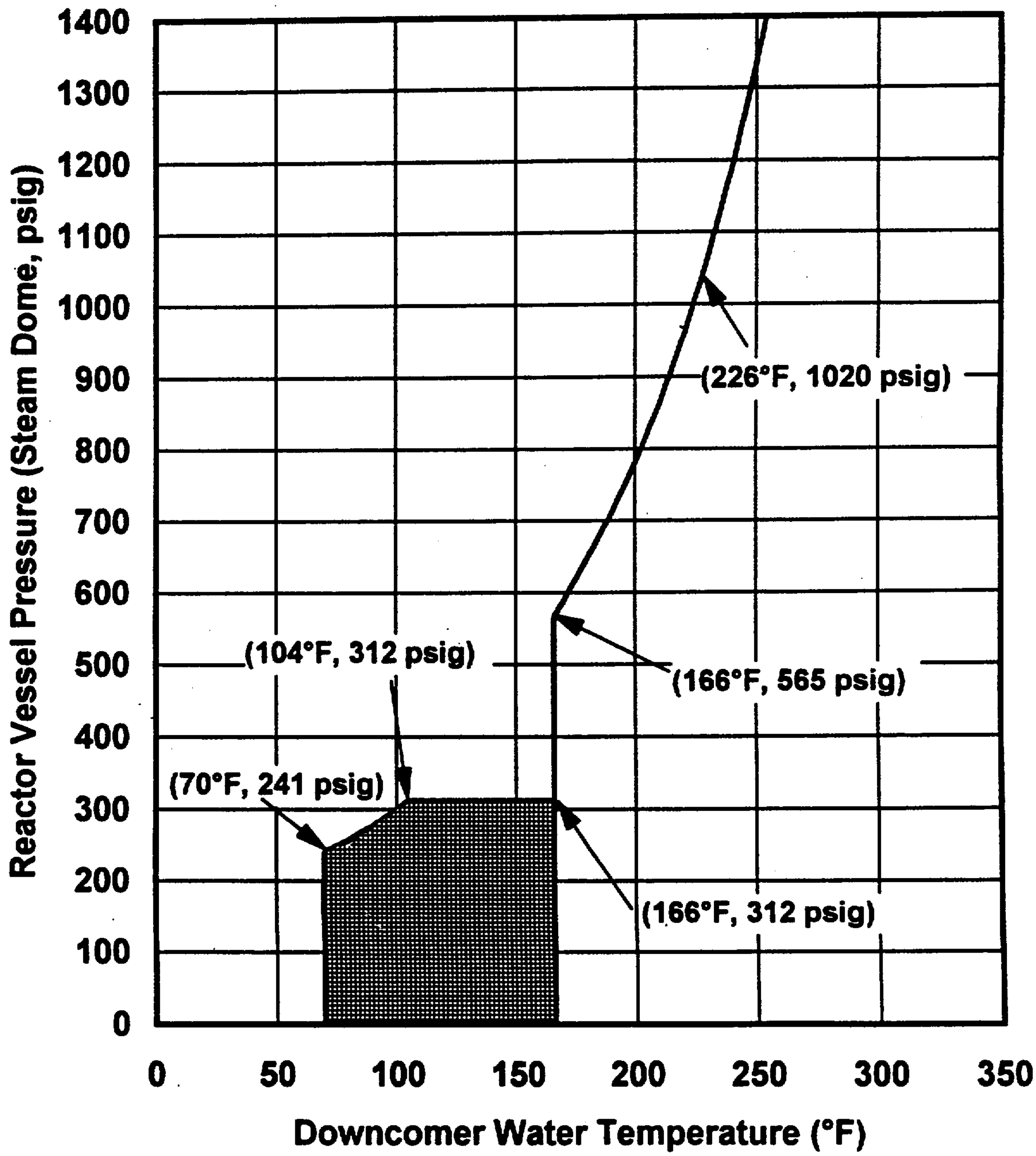

Figure 3.4.11-5 (Page 1 of 1)
Nuclear Cooldown Curve

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Reactor Steam Dome Pressure

LCO 3.4.12 The reactor steam dome pressure shall be ≤ 1035 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.12.1 Verify reactor steam dome pressure is ≤ 1035 psig.	12 hours

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS—Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1	Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. High Pressure Core Spray (HPCS) System inoperable.	B.1	Verify by administrative means RCIC System is OPERABLE when RCIC is required to be OPERABLE.	Immediately
	AND		
	B.2	Restore HPCS System to OPERABLE status.	14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two ECCS injection subsystems inoperable. OR One ECCS injection and one ECCS spray subsystem inoperable.	C.1 Restore one ECCS injection/spray subsystem to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours
E. One required ADS valve inoperable.	E.1 Restore ADS valve to OPERABLE status.	14 days
F. One required ADS valve inoperable. AND One low pressure ECCS injection/spray subsystem inoperable.	F.1 Restore ADS valve to OPERABLE status. OR F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. Required Action and associated Completion Time of Condition E or F not met. OR Two or more required ADS valves inoperable.	G.1	Be in MODE 3.	12 hours
	AND		
	G.2	Reduce reactor steam dome pressure to ≤ 150 psig.	36 hours
H. HPCS and Low Pressure Core Spray (LPCS) Systems inoperable. OR Three or more ECCS injection/spray subsystems inoperable. OR HPCS System and one or more required ADS valves inoperable. OR Two or more ECCS injection/spray subsystems and one or more required ADS valves inoperable.	H.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.1.2	-------------------NOTE------------------- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the residual heat removal cut-in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. -- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.1.3	Verify: a. For each ADS nitrogen receiver discharge header, the pressure is ≥ 160 psig; and b. For each ADS nitrogen receiver tank, the pressure is ≥ 334 psig.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.1.4	Verify each ECCS pump develops the specified flow rate with the specified developed head. SYSTEM / FLOW RATE / TOTAL DEVELOPED HEAD LPCS / ≥ 6350 gpm / ≥ 284 psid LPCI A, B / ≥ 7450 gpm / ≥ 127 psid LPCI C / ≥ 7450 gpm / ≥ 140 psid HPCS / ≥ 6350 gpm / ≥ 327 psid	In accordance with the Inservice Testing Program
SR 3.5.1.5	-------------------NOTE------------------- Vessel injection/spray may be excluded. -- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.6	-------------------NOTE------------------- Valve actuation may be excluded. -- Verify the ADS actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.7	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each required ADS valve actuator strokes when manually actuated.	24 months on a STAGGERED TEST BASIS for each valve solenoid

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.8	-------------------NOTE------------------- Instrumentation response time may be assumed to be the design instrumentation response time. -- Verify the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is within limits.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.2 ECCS—Shutdown

LCO 3.5.2 Two ECCS injection/spray subsystems shall be OPERABLE.

APPLICABILITY: MODE 4,
MODE 5 except with the spent fuel storage pool gates removed and water level ≥ 22 ft 3 inches over the top of the reactor pressure vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ECCS injection/spray subsystem inoperable.	A.1 Restore required ECCS injection/spray subsystem to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
C. Two required ECCS injection/spray subsystems inoperable.	C.1 Initiate action to suspend OPDRVs.	Immediately
	<u>AND</u>	
	C.2 Restore one ECCS injection/spray subsystem to OPERABLE status.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action C.2 and associated Completion Time not met.	D.1 Initiate action to restore secondary containment to OPERABLE status.	Immediately
	AND	
	D.2 Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
	AND	
	D.3 Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify, for each required low pressure ECCS injection/spray subsystem, the suppression pool water level is ≥ 195 ft.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.2	Verify, for the required High Pressure Core Spray (HPCS) System, the: a. Suppression pool water level is $\geq$ 195 ft; or b. Condensate storage tank B water level is $\geq$ 26.9 ft.	12 hours
SR 3.5.2.3	Verify, for each required ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.2.4	-------------------NOTE------------------- One low pressure coolant injection (LPCI) subsystem may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned and not otherwise inoperable. -- Verify each required ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.5	Verify each required ECCS pump develops the specified flow rate with the specified developed head. SYSTEM / FLOW RATE / TOTAL DEVELOPED HEAD LPCS ≥ 6350 gpm ≥ 284 psid LPCI A, B ≥ 7450 gpm ≥ 127 psid LPCI C ≥ 7450 gpm ≥ 140 psid HPCS ≥ 6350 gpm ≥ 327 psid	In accordance with the Inservice Testing Program
SR 3.5.2.6	-------------------NOTE------------------- Vessel injection/spray may be excluded. -- Verify each required ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.2.7	-------------------NOTE------------------- Instrumentation response time may be assumed to be the design instrumentation response time. -- Verify the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is within limits.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE. AND A.2 Restore RCIC System to OPERABLE status.	Immediately 14 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 1035 psig and ≥ 935 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.3.5	-------------------NOTE------------------- Vessel injection may be excluded. -- Verify the RCIC System actuates on an actual or simulated automatic initiation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.1.1 Primary Containment

LCO 3.6.1.1 Primary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Primary containment inoperable.	A.1 Restore primary containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

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 10 CFR 50 Appendix J Testing Program Plan.	In accordance with 10 CFR 50 Appendix J Testing Program Plan
SR 3.6.1.1.2	Verify the drywell-to-suppression chamber bypass leakage rate is less than or equal to the equivalent leakage rate through a orifice 0.0054 ft^2 at an initial differential pressure of ≥ 3 psid.	In accordance with the Type A testing frequency of the 10 CFR 50 Appendix J Testing Program Plan AND -----NOTE------ Only required after two consecutive tests fail and continues until two consecutive tests pass -------------- 24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.1.3	-------------------NOTE------------------- SR 3.6.1.1.2 may be performed in lieu of SR 3.6.1.1.3. -- Verify, at an initial differential pressure of ≥ 3 psid: a. The leakage rate through each drywell-to-suppression chamber bypass leak path containing suppression chamber-to-drywell vacuum breakers is less than or equal to the equivalent through an orifice 0.000648 ft^2; and b. The combined leakage rate through all four drywell-to-suppression chamber bypass leak paths containing suppression chamber-to-drywell vacuum breakers is less than or equal to the equivalent through an orifice 0.001296 ft^2.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.1.2 Primary Containment Air Locks

LCO 3.6.1.2 Two primary containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

1. Entry and exit is permissible to perform repairs of the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when air lock leakage results in exceeding overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more primary containment air locks with one primary containment air lock door inoperable.	------------NOTES------------ 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. -----------------------------	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Verify the OPERABLE door is closed in the affected air lock.	1 hour
	AND	
	A.2 Lock the OPERABLE door closed in the affected air lock.	24 hours
	AND	
	A.3 --------NOTE--------- Air lock doors in high radiation areas or areas with limited access due to inerting may be verified locked closed by administrative means. --------------------	
	Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days
B. One or more primary containment air locks with primary containment air lock interlock mechanism inoperable.	------------NOTES------------ 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry into and exit from primary containment is permissible under the control of a dedicated individual. -----------------------------	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.1	Verify an OPERABLE door is closed in the affected air lock.	1 hour
	<u>AND</u>		
	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours
	<u>AND</u>		
	B.3	--------NOTE--------- Air lock doors in high radiation areas or areas with limited access due to inerting may be verified locked closed by administrative means. ---------------------	
		Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more primary containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate primary containment overall leakage rate per LCO 3.6.1.1, using current air lock test results.	Immediately
	<u>AND</u>		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	<u>AND</u>		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	D.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.2.1	-------------------NOTES------------------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.1. --- Perform required primary containment air lock leakage rate testing in accordance with 10 CFR 50 Appendix J Testing Program Plan.	In accordance with 10 CFR 50 Appendix J Testing Program Plan
SR 3.6.1.2.2	Verify only one door in the primary containment air lock can be opened at a time.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV and each non-PCIV listed in Table 3.6.1.3-1 shall be OPERABLE.

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

ACTIONS

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

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------NOTE--------- Only applicable to penetration flow paths with two or more PCIVs. ---------------------- One or more penetration flow paths with one PCIV inoperable except due to leakage not within limit.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours except for main steam line <u>AND</u> 8 hours for main steam line
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	--------NOTES-------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. -------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside primary containment AND Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ---------NOTE--------- Only applicable to penetration flow paths with two or more PCIVs. --------------------- One or more penetration flow paths with two or more PCIVs inoperable except due to leakage not within limit.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. ---------NOTE--------- Only applicable to penetration flow paths with only one PCIV. --------------------- One or more penetration flow paths with one PCIV inoperable except due to leakage not within limit.	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	4 hours except for excess flow check valves (EFCVs) and penetrations with a closed system <u>AND</u> 72 hours for EFCVs and penetrations with a closed system
	<u>AND</u>		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2	---------NOTES--------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more penetration flowpaths with secondary containment bypass leakage rate, MSIV leakage rate, or hydrostatically tested line leakage rate not within limit.	D.1	--------NOTE--------- The isolation device used to satisfy Required Action D.1 shall have been verified to meet the applicable leakage rate limit of the inoperable valve. --------------------- Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	4 hours for hydrostatically tested line leakage not on a closed system AND 4 hours for secondary containment bypass leakage AND 8 hours for MSIV leakage AND 72 hours for hydrostatically tested line leakage on a closed system
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	--------NOTES-------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside primary containment AND Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.3	Perform SR 3.6.1.3.6 for the resilient seal purge supply valves closed to comply with Required Action D.1.	Once per 92 days
E. One or more penetration flow paths with one or more containment purge exhaust valves not within purge valve leakage limits.	E.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	24 hours
	<u>AND</u>		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.2 ---------NOTES--------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside containment
	AND	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.3 Perform SR 3.6.1.3.6 for the resilient seal purge exhaust valves closed to comply with Required Action E.1.	Once per 92 days
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met in MODE 1, 2, or 3.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	12 hours 36 hours
G. Required Action and associated Completion Time of Condition A, B, C, D, or E not met for PCIV(s) required to be OPERABLE during MODE 4 or 5.	G.1 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRV)s. OR G.2 Initiate action to restore valve(s) to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.1	-------------------NOTE------------------- Not required to be met when the 12 inch and 14 inch primary containment purge valves are open for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open, provided: a) the Standby Gas Treatment (SGT) System is OPERABLE; or b) the primary containment full flow line to the SGT System is isolated and one SGT subsystem is OPERABLE. -- Verify each 12 inch and 14 inch primary containment purge valve is closed.	31 days
SR 3.6.1.3.2	-------------------NOTES------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. -- Verify each primary containment isolation manual valve and blind flange that is located outside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.3	-------------------NOTES------------------ 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. --- Verify each primary containment isolation manual valve and blind flange that is located inside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4, if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except MSIVs, is within limits.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Perform leakage rate testing for each primary containment purge valve with resilient seals.	184 days AND Once within 92 days after opening the valve
SR 3.6.1.3.7	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.8	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.9	Verify each EFCV actuates to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.10	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.11	Verify the leakage rate for the secondary containment bypass leakage paths is within the limits of Table 3.6.1.3-1 when pressurized to ≥ 40 psig.	In accordance with 10 CFR 50 Appendix J Testing Program Plan

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.12	Verify leakage rate through each MSIV is ≤ 24 scfh when tested at ≥ 40 psig.	In accordance with 10 CFR 50 Appendix J Testing Program Plan
SR 3.6.1.3.13	Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with 10 CFR 50 Appendix J Testing Program Plan

Table 3.6.1.3-1 (page 1 of 2)
Secondary Containment Bypass Leakage Paths Leakage Rate Limits

VALVE NUMBER	PER VALVE LEAK RATE (SCFH)
2MSS*MOV111 2MSS*MOV112	1.875
2MSS*MOV208	0.625
2CMS*SOV74A, B 2CMS*SOV75A, B 2CMS*SOV76A, B 2CMS*SOV77A, B	0.2344
2DER*MOV119 2DER*RV344	(a)
2DER*MOV120	1.25
2DER*MOV130 2DER*MOV131	0.625
2DFR*MOV120	1.875
2DFR*MOV121 2DFR*RV228	(b)
2DFR*MOV139 2DFR*MOV140	0.9375
2WCS*MOV102 2WCS*MOV112	2.5
2FWS*AOV23A, B 2FWS*V12A, B	12.0
2CPS*AOV104 2CPS*AOV106	4.38

(continued)

(a) The combined leakage rate for these two valves shall be ≤ 1.25 SCFH.

(b) The combined leakage rate for these two valves shall be ≤ 1.875 SCFH.

Table 3.6.1.3-1 (page 2 of 2)
Secondary Containment Bypass Leakage Paths Leakage Rate Limits

VALVE NUMBER	PER VALVE LEAK RATE (SCFH)
2CPS*AOV105 2CPS*AOV107	3.75
2CPS*SOV119 2CPS*SOV120 2CPS*SOV121 2CPS*SOV122	0.625
2IAS*SOV164 2IAS*V448	0.9375
2IAS*SOV165 2IAS*V449	0.9375
2GSN*SOV166 2GSN*V170	(c)
2IAS*SOV166 2IAS*SOV184	(c)
2IAS*SOV167 2IAS*SOV185	(c)
2IAS*SOV168 2IAS*SOV180	(c)
2CPS*SOV132 2CPS*V50	(c)
2CPS*SOV133 2CPS*V51	(c)

(c) The combined leak rate for these penetrations shall be ≤ 3.6 SCFH. The assigned leakage rate through a penetration shall be that of the valve with the highest leakage rate in that penetration. However, if a penetration is isolated by one closed and de-activated automatic valve, closed manual valve, or blind flange, the leakage through the penetration shall be the actual pathway leakage.

3.6 CONTAINMENT SYSTEMS

3.6.1.4 Drywell and Suppression Chamber Pressure

LCO 3.6.1.4 Drywell and suppression chamber pressure shall be $\geq$ 14.2 psia and $\leq$ 15.45 psia.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell or suppression chamber pressure not within limits.	A.1 Restore drywell and suppression chamber pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.4.1 Verify drywell and suppression chamber pressure is within limits.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.1.5 Drywell Air Temperature

LCO 3.6.1.5 Drywell average air temperature shall be ≤ 150°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell average air temperature not within limit.	A.1 Restore drywell average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.5.1 Verify drywell average air temperature is within limit.	24 hours

3.6 CONTAINMENT SYSTEMS

3.6.1.6 Residual Heat Removal (RHR) Drywell Spray System

LCO 3.6.1.6 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1	Restore RHR drywell spray subsystem to OPERABLE status.	7 days
B. Two RHR drywell spray subsystems inoperable.	B.1	Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.6.1	Verify each RHR drywell spray subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.1.6.2	Verify, by administrative means, that each required RHR pump is OPERABLE.	92 days
SR 3.6.1.6.3	Verify each drywell spray nozzle is unobstructed.	10 years

3.6 CONTAINMENT SYSTEMS

3.6.1.7 Suppression Chamber-to-Drywell Vacuum Breakers

LCO 3.6.1.7 Each suppression chamber-to-drywell vacuum breaker shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One line with one or more suppression chamber-to-drywell vacuum breakers inoperable for opening.	A.1 Restore the vacuum breaker(s) to OPERABLE status.	72 hours
B. ---------NOTE--------- Separate Condition entry is allowed for each suppression chamber-to-drywell vacuum breaker line. --------------------- One or more lines with one suppression chamber-to-drywell vacuum breaker not closed.	B.1 Close the open vacuum breaker.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE---------- Separate Condition entry is allowed for each suppression chamber-to-drywell vacuum breaker line. --------------------- One or more lines with two suppression chamber-to-drywell vacuum breakers disks not closed.	C.1 Close one open vacuum breaker.	2 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	12 hours
	AND	
	D.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.7.1	------------------NOTES------------------ 1. Not required to be met for vacuum breakers that are open during Surveillances. 2. Not required to be met for vacuum breakers open when performing their intended function. --- Verify each vacuum breaker is closed.	14 days
SR 3.6.1.7.2	Perform a functional test of each vacuum breaker.	31 days <u>AND</u> Within 12 hours after any discharge of steam to the suppression chamber from the safety/relief valves
SR 3.6.1.7.3	Verify the opening setpoint of each vacuum breaker is $\leq$ 0.25 psid.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.2.1 Suppression Pool Average Temperature

LCO 3.6.2.1 Suppression pool average temperature shall be:

a. ≤ 90°F with THERMAL POWER > 1% RTP and no testing that adds heat to the suppression pool is being performed;

b. ≤ 105°F with THERMAL POWER > 1% RTP and testing that adds heat to the suppression pool is being performed; and

c. ≤ 110°F with THERMAL POWER ≤ 1% RTP.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool average temperature > 90°F but ≤ 110°F. AND THERMAL POWER > 1% RTP. AND Not performing testing that adds heat to the suppression pool.	A.1 Verify suppression pool average temperature ≤ 110°F. AND A.2 Restore suppression pool average temperature to ≤ 90°F.	Once per hour 24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to ≤ 1% RTP.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Suppression pool average temperature > 105°F. AND THERMAL POWER > 1% RTP. AND Performing testing that adds heat to the suppression pool.	C.1 Suspend all testing that adds heat to the suppression pool.	Immediately
D. Suppression pool average temperature > 110°F but ≤ 120°F.	D.1 Place the reactor mode switch in the shutdown position. AND D.2 Verify suppression pool average temperature ≤ 120°F. AND D.3 Be in MODE 4.	Immediately Once per 30 minutes 36 hours
E. Suppression pool average temperature > 120°F.	E.1 Depressurize the reactor vessel to < 200 psig. AND E.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1.1	Verify suppression pool average temperature is within the applicable limits.	24 hours AND 5 minutes when performing testing that adds heat to the suppression pool

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Suppression pool water level shall be ≥ 199 ft 6 inches and ≤ 201 ft

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool water level not within limits.	A.1 Restore suppression pool water level to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.2.2.1 Verify suppression pool water level is within limits.	24 hours

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days
B. Two RHR suppression pool cooling subsystems inoperable.	B.1 Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool 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.	31 days
SR 3.6.2.3.2	Verify each required RHR pump develops a flow rate $\geq$ 7450 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	In accordance with the Inservice Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

LCO 3.6.2.4 Two RHR suppression pool spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR suppression pool spray subsystem inoperable.	A.1	Restore RHR suppression pool spray subsystem to OPERABLE status.	7 days
B. Two RHR suppression pool spray subsystems inoperable.	B.1	Restore one RHR suppression pool spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify each RHR suppression pool spray subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.2.4.2	Verify each required RHR pump develops a flow rate $\geq$ 450 gpm while operating in the suppression pool spray mode.	In accordance with the Inservice Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.3.1 Primary Containment Hydrogen Recombiners

LCO 3.6.3.1 Two primary containment hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One primary containment hydrogen recombiner inoperable.	A.1 --------NOTE--------- LCO 3.0.4 is not applicable. --------------------- Restore primary containment hydrogen recombiner to OPERABLE status.	30 days
B. Two primary containment hydrogen recombiners inoperable.	B.1 Verify by administrative means that the hydrogen and oxygen control function is maintained.	1 hour <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>	
	B.2 Restore one primary containment hydrogen recombiner to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1.1	Perform a system functional test for each primary containment hydrogen recombiner.	24 months
SR 3.6.3.1.2	Perform a resistance to ground test for each heater phase.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.3.2 Primary Containment Oxygen Concentration

LCO 3.6.3.2 The primary containment oxygen concentration shall be < 4.0 volume percent.

APPLICABILITY: MODE 1 during the time period:

a. From 24 hours after THERMAL POWER is > 15% RTP following startup, to

b. 24 hours prior to reducing THERMAL POWER to < 15% RTP prior to the next scheduled reactor shutdown.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Primary containment oxygen concentration not within limit.	A.1 Restore oxygen concentration to within limit.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 15% RTP.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.3.2.1 Verify primary containment oxygen concentration is within limits.	7 days

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Secondary containment inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1	---------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>		
	C.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	C.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is $\geq$ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.3	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.4	Verify the secondary containment can be drawn down to $\geq$ 0.25 inch of vacuum water gauge in $\leq$ 66.7 seconds using one standby gas treatment (SGT) subsystem.	24 months on a STAGGERED TEST BASIS for each SGT subsystem
SR 3.6.4.1.5	Verify the secondary containment can be maintained $\geq$ 0.25 inch of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate $\leq$ 2670 cfm.	24 months on a STAGGERED TEST BASIS for each SGT subsystem

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

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

ACTIONS

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

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	8 hours
	<u>AND</u>		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	--------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. ---------------------	
		Verify the affected penetration flow path is isolated.	Once per 31 days
B. ---------NOTE--------- Only applicable to penetration flow paths with two isolation valves. ---------------------- One or more penetration flow paths with two SCIVs inoperable.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	D.1	--------NOTE-------- LCO 3.0.3 is not applicable. -------------------- Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>		
	D.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	D.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.2.1	------------------NOTES------------------ 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for SCIVs that are open under administrative controls. --- Verify 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.	31 days
SR 3.6.4.2.2	Verify the isolation time of each power operated, automatic SCIV is within limits.	92 days
SR 3.6.4.2.3	Verify each automatic SCIV actuates to the isolation position on an actual or simulated automatic isolation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- C.1 Place OPERABLE SGT subsystem in operation. OR	 Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment. AND	Immediately
	C.2.2 Suspend CORE ALTERATIONS. AND	Immediately
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately
E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1 ---------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of irradiated fuel assemblies in the secondary containment. AND	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.2 Suspend CORE ALTERATIONS. <u>AND</u> E.3 Initiate action to suspend OPDRVs.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.6.4.3.4	Verify each SGT decay heat removal air inlet valve can be opened.	24 months

3.7 PLANT SYSTEMS

3.7.1 Service Water (SW) System and Ultimate Heat Sink (UHS)

LCO 3.7.1 Division 1 and 2 SW subsystems and UHS shall be OPERABLE.

AND

Four OPERABLE SW pumps shall be in operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SW supply header cross connect valve inoperable.	A.1	Open the SW supply header cross connect valve.	1 hour
	AND		
	A.2	Restore the SW supply header cross connect valve to OPERABLE status.	72 hours
B. One or more non-safety related SW flow paths with one SW isolation valve inoperable.	B.1	Isolate the affected non-safety related SW flow path(s).	72 hours
C. One SW subsystem inoperable for reasons other than Conditions A and B.	C.1	Restore SW subsystem to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One division of intake deicer heaters inoperable.	D.1 Restore intake deicer heater division to OPERABLE status.	72 hours
E. One required SW pump not in operation.	E.1 Restore required SW pump to operation.	72 hours
F. Two or more required SW pumps not in operation.	F.1 Restore all but one required SW pump to operation.	1 hour
G. Required Action and associated Completion Time of Condition A, B, C, D, E, or F not met. OR Both SW subsystems inoperable for reasons other than Conditions A, B, and C. OR UHS inoperable for reasons other than Condition D.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," for RHR Shutdown Cooling subsystem(s) made inoperable by SW System or UHS. ---------------------------- G.1 Be in MODE 3. AND G.2 Be in MODE 4.	 12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-------------------NOTE------------------- Not required to be met if SR 3.7.1.5 and SR 3.7.1.8 satisfied. -- Verify the water temperature of the intake tunnels is ≥ 38°F.	12 hours
SR 3.7.1.2	Verify the water level in the SW pump intake bay is ≥ 233.1 ft.	24 hours
SR 3.7.1.3	Verify the water temperature of each SW subsystem supply header is ≤ 82°F.	24 hours <u>AND</u> 4 hours when supply header water temperature is ≥ 75°F <u>AND</u> 2 hours when supply header water temperature is ≥ 79°F
SR 3.7.1.4	Verify each required SW pump is in operation.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.1.5	-------------------NOTE------------------- Not required to be met if SR 3.7.1.1 satisfied. -- Verify, for each intake deicer heater division, the current of each required heater feeder cable is within the limit.	7 days
SR 3.7.1.6	-------------------NOTE------------------- Isolation of flow to individual components does not render SW System inoperable. -- Verify each SW subsystem manual, power operated, and automatic valve in the flow path servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.1.7	Verify each SW subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.7.1.8	-------------------NOTE------------------- Not required to be met if SR 3.7.1.1 satisfied. -- Verify, for each intake deicer heater division, the resistance of each required heater feeder cable and associated heater elements is within the limit.	24 months

3.7 PLANT SYSTEMS

3.7.2 Control Room Envelope Filtration (CREF) System

LCO 3.7.2 Two CREF subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREF subsystem inoperable. <u>OR</u> Two CREF subsystems inoperable with safety function maintained.	A.1 Restore CREF subsystem(s) to OPERABLE status.	7 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	C.1 Place OPERABLE components of CREF subsystem(s) equivalent to a single CREF subsystem in emergency pressurization mode.	Immediately
	<u>OR</u>	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two CREF subsystems inoperable with safety function not maintained in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CREF subsystems inoperable with safety function not maintained during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. -----------------------------	
	E.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Operate each CREF subsystem for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.2.2	Perform required CREF System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.2.3	Verify each CREF subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.7.2.4	Verify all combinations of the CREF System can maintain a positive pressure of $\geq$ 1/8 inches water gauge relative to outside atmosphere during the emergency pressurization mode of operation at an outside air intake flow rate of $\leq$ 1500 cfm.	24 months

3.7 PLANT SYSTEMS

3.7.3 Control Room Envelope Air Conditioning (AC) System

LCO 3.7.3 Two control room envelope AC subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room envelope AC subsystem inoperable. OR Two control room envelope AC subsystems inoperable with safety function maintained.	A.1 Restore control room envelope AC subsystem(s) to OPERABLE status.	30 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	C.1 Place OPERABLE components of control room envelope AC subsystem(s) equivalent to a single control room envelope AC subsystem in operation.	Immediately
	<u>OR</u>	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two control room envelope AC subsystems inoperable with safety function not maintained in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two control room envelope AC subsystems inoperable with safety function not maintained during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	E.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify each control room envelope AC subsystem has the capability to remove the assumed heat load.	24 months

3.7 PLANT SYSTEMS

3.7.4 Main Condenser Offgas

LCO 3.7.4 The gross gamma activity rate of the noble gases measured at the offgas recombiner effluent shall be ≤ 350,000 μCi/second after decay of 30 minutes.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated and steam jet air ejector (SJAE) in operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gross gamma activity rate of the noble gases not within limit.	A.1 Restore gross gamma activity rate of the noble gases to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Isolate all main steam lines.	12 hours
	<u>OR</u>	
	B.2 Isolate SJAE.	12 hours
	<u>OR</u>	
	B.3.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	B.3.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	-------------------NOTE-------------------- Not required to be performed until 31 days after any main steam line not isolated and SJAE in operation. --- Verify the gross gamma activity rate of the noble gases is ≤ 350,000 μCi/second after decay of 30 minutes.	31 days <u>AND</u> Once within 4 hours after a ≥ 50% increase in the nominal steady state fission gas release after factoring out increases due to changes in THERMAL POWER level

3.7 PLANT SYSTEMS

3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Perform a system functional test.	24 months
SR 3.7.5.2	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

3.7 PLANT SYSTEMS

3.7.6 Spent Fuel Storage Pool Water Level

LCO 3.7.6 The spent fuel storage pool water level shall be ≥ 22 ft 3 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool,
During movement of new fuel assemblies in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of fuel assemblies in the spent fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the spent fuel storage pool water level is ≥ 22 ft 3 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electric Power Distribution System; and

b. Three diesel generators (DGs).

APPLICABILITY: MODES 1, 2, and 3.

----------------------------NOTE----------------------------
Division 3 AC electrical power sources are not required to be OPERABLE when High Pressure Core Spray (HPCS) System is inoperable.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one division concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	A.3	Restore required offsite circuit to OPERABLE status.	72 hours <u>AND</u> 24 hours from discovery of both HPCS and Low Pressure Core Spray (LPCS) Systems with no offsite power <u>AND</u> 6 days from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One required DG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	AND		
	B.4	Restore required DG to OPERABLE status.	72 hours AND 6 days from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	<u>AND</u>	
	C.2 Restore one required offsite circuit to OPERABLE status.	24 hours
D. One required offsite circuit inoperable. <u>AND</u> One required DG inoperable.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.8, "Distribution Systems—Operating," when Condition D is entered with no AC power source to any division. ----------------------------	
	D.1 Restore required offsite circuit to OPERABLE status.	12 hours
	<u>OR</u>	
	D.2 Restore required DG to OPERABLE status.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two required DGs inoperable.	E.1 Restore one required DG to OPERABLE status.	2 hours <u>OR</u> 24 hours if Division 3 DG is inoperable
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	12 hours 36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.1.1 Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.2	-------------------NOTE------------------- All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. -- Verify each required DG starts from standby conditions and achieves: a. In $\leq$ 10 seconds, voltage $\geq$ 3950 V for Division 1 and 2 DGs and $\geq$ 3820 V for Division 3 DG, and frequency $\geq$ 58.8 Hz for Division 1 and 2 DGs and $\geq$ 58.0 Hz for Division 3 DG; and b. Steady state voltage $\geq$ 3950 V and $\leq$ 4370 V and frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.3	------------------NOTES------------------ 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by, and immediately follow, without shutdown, a successful performance of SR 3.8.1.2. -- Verify each required DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 3960 kW and ≤ 4400 kW for Division 1 and 2 DGs, and ≥ 2340 kW and ≤ 2600 kW for Division 3 DG.	31 days
SR 3.8.1.4	Verify each required day tank contains ≥ 403 gal of fuel oil for Division 1 and 2 DGs and ≥ 282 gal for Division 3 DG.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each required day tank.	31 days
SR 3.8.1.6	Verify each required fuel oil transfer subsystem operates to automatically transfer fuel oil from the storage tank to the day tank.	62 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.7	-------------------NOTES------------------ 1. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. 2. If performed with DG synchronized with offsite power, it shall be performed within the power factor limit. However if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. -- Verify each required DG rejects a load greater than or equal to its associated single largest post-accident load, and following load rejection, the frequency is ≤ 64.5 Hz for Division 1 and 2 DGs and ≤ 66.75 Hz for Division 3 DG.	 24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	-------------------NOTES------------------ 1. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. 2. If grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. -- Verify each required DG operating within the power factor limit does not trip and voltage is maintained: a. ≤ 4576 V during and following a load rejection of a load ≥ 4400 kW for Division 1 and 2 DGs; and b. ≤ 5824 V during and following a load rejection of a load ≥ 2600 kW for Division 3 DG.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	-------------------NOTES------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. --- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses for Divisions 1 and 2 only; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 13.20 seconds, 2. energizes auto-connected shutdown loads for Division 1 and 2 DGs only, through the associated automatic load sequence time delay relays, 3. maintains steady state voltage ≥ 3950 V and ≤ 4370 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected shutdown loads for ≥ 5 minutes for Division 1 and 2 DGs and supplies permanently connected shutdown loads for ≥ 5 minutes for Division 3 DG.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	-------------------NOTES------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. ---	
	Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal each required DG auto-starts from standby condition and: a. In ≤ 10 seconds after auto-start, achieves voltage ≥ 3950 V for Division 1 and 2 DGs and ≥ 3820 V for Division 3 DG, and frequency ≥ 58.8 Hz for Division 1 and 2 DGs and ≥ 58.0 Hz for Division 3 DG; b. Achieves steady state voltage ≥ 3950 V and ≤ 4370 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz; c. Operates for ≥ 5 minutes; d. Permanently connected loads remain energized from the offsite power system for Divisions 1 and 2 only; and e. Emergency loads are auto-connected through the associated automatic load sequence time delay relays to the offsite power system for Divisions 1 and 2 only.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. -- Verify each required DG's automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ECCS initiation signal except: a. Engine overspeed; and b. Generator differential current.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-------------------NOTES------------------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not be performed in MODE 1 or 2 unless the other two DGs are OPERABLE. If either of the other two DGs become inoperable, this Surveillance shall be suspended. However, credit may be taken for unplanned events that satisfy this SR. 3. If grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ---	
	Verify each required DG operating within the power factor limit operates for ≥ 24 hours: a. For ≥ 2 hours loaded ≥ 4620 kW and ≤ 4840 kW for Division 1 and 2 DGs, and ≥ 2730 kW and ≤ 2860 kW for Division 3 DG; and b. For the remaining hours of the test loaded ≥ 3960 kW and ≤ 4400 kW for Division 1 and 2 DGs, and ≥ 2340 kW and ≤ 2600 kW for Division 3 DG.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	-------------------NOTES------------------- 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 2 hours loaded ≥ 3960 kW for Division 1 and 2 DGs, and ≥ 2340 kW for Division 3 DG. Momentary transients below the load limit do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. --- Verify each required DG starts and achieves: a. In ≤ 10 seconds, voltage ≥ 3950 V for Division 1 and 2 DGs and ≥ 3820 V for Division 3 DG, and frequency ≥ 58.8 Hz for Division 1 and 2 DGs and ≥ 58.0 Hz for Division 3 DG; and b. Steady state voltage ≥ 3950 V and ≤ 4370 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. -- Verify each required DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	24 months
SR 3.8.1.15	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. -- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ECCS initiation signal overrides the test mode by: a. Returning DG to ready-to-load operation; and b. Automatically energizing the emergency load from offsite power.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.16	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. -- Verify interval between each sequenced load block, for the Division 1 and 2 DGs only, is ≥ 90% of the design interval for each automatic load sequence time delay relay.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.17	-------------------NOTES------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. --- Verify, on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses for Divisions 1 and 2 only; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. for Divisions 1 and 2, energizes auto-connected emergency loads through the associated automatic load sequence time delay relays and for Division 3, energizes auto-connected emergency loads, 3. maintains steady state voltage ≥ 3950 V and ≤ 4370 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.18	------------------NOTE------------------ All DG starts may be preceded by an engine prelube period. -- Verify, when started simultaneously from standby condition, each Division 1, 2, and 3 DG achieves, in ≤ 10 seconds, voltage ≥ 3950 V for Division 1 and 2 DGs and ≥ 3820 V for Division 3 DG, and frequency ≥ 58.8 Hz for Division 1 and 2 DGs and ≥ 58.0 Hz for Division 3 DG.	10 years

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources—Shutdown

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

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.9, "Distribution Systems—Shutdown"; and

b. One diesel generator (DG) capable of supplying one division of the Division 1 or 2 onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.9; and

c. One qualified circuit, other than the circuit in LCO 3.8.2.a, between the offsite transmission and the Division 3 onsite Class 1E electrical power distribution subsystem, or the Division 3 DG capable of supplying the Division 3 onsite Class 1E AC electrical power distribution subsystem, when the Division 3 onsite Class 1E electrical power distribution subsystem is required by LCO 3.8.9.

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

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO Item a. not met.	------------NOTE------------ Enter applicable Condition and Required Actions of LCO 3.8.9, when any required division is de-energized as a result of Condition A. ----------------------------	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. LCO Item b. not met.	B.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	B.2 Suspend movement of irradiated fuel assemblies in secondary containment.	Immediately
	<u>AND</u>	
	B.3 Initiate action to suspend OPDRVs.	Immediately
	<u>AND</u>	
	B.4 Initiate action to restore required DG to OPERABLE status.	Immediately
C. LCO Item c. not met.	C.1 Declare High Pressure Core Spray System inoperable.	72 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.2.1	-------------------NOTES------------------- 1. The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.7 through SR 3.8.1.9, SR 3.8.1.11 through SR 3.8.1.14, SR 3.8.1.16, and SR 3.8.1.17. 2. SR 3.8.1.10 and SR 3.8.1.17 are not required to be met when associated ECCS subsystem(s) are not required to be OPERABLE per LCO 3.5.2, "ECCS—Shutdown." --- For AC sources required to be OPERABLE, the SRs of Specification 3.8.1, except SR 3.8.1.15 and SR 3.8.1.18, are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with stored fuel oil level: 1. For Division 1 DG or Division 2 DG, < 50,000 gal and ≥ 44,000 gal; and 2. For Division 3 DG, < 35,342 gal and ≥ 30,813 gal.	A.1 Restore stored fuel oil level to within limit.	48 hours
B. One or more DGs with lube oil inventory: 1. For Division 1 DG or Division 2 DG, < 99 gal and ≥ 84 gal; and 2. For Division 3 DG, < 168 gal and ≥ 144 gal.	B.1 Restore lube oil inventory to within limit.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more DGs with stored fuel oil total particulates not within limit.	C.1	Restore stored fuel oil total particulates to within limit.	7 days
D. One or more DGs with new fuel oil properties not within limits.	D.1	Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with starting air receiver pressure: 1. For Division 1 DG or Division 2 DG, < 225 psig and ≥ 175 psig; and 2. For Division 3 DG, < 190 psig and ≥ 110 psig.	E.1	Restore starting air receiver pressure to within limit.	48 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. <u>OR</u> One or more DGs with stored diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1	Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains: a. ≥ 50,000 gal of fuel for Division 1 DG and Division 2 DG; and b. ≥ 35,342 gal of fuel for Division 3 DG.	31 days
SR 3.8.3.2	Verify lube oil inventory is: a. ≥ 99 gal for Division 1 DG and Division 2 DG; and b. ≥ 168 gal for Division 3 DG.	31 days
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG air start receiver pressure is: a. ≥ 225 psig for Division 1 DG and Division 2 DG; and b. ≥ 190 psig for Division 3 DG.	31 days
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources—Operating

LCO 3.8.4 The Division 1, Division 2, and Division 3 DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Division 1 or 2 DC electrical power subsystem inoperable.	A.1	Restore Division 1 and 2 DC electrical power subsystems to OPERABLE status.	2 hours
B. Division 3 DC electrical power subsystem inoperable.	B.1	Declare High Pressure Core Spray System inoperable.	Immediately
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 130 V on float charge.	7 days
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. <u>OR</u> Verify battery connection resistance is ≤ 20% above the resistance as measured during installation for intercell and terminal connections.	92 days
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	24 months
SR 3.8.4.4	Remove visible corrosion and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	24 months
SR 3.8.4.5	Verify battery connection resistance is ≤ 20% above the resistance as measured during installation for intercell and terminal connections.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.4.6	Verify each required Division 1 and 2 battery charger supplies ≥ 300 amps and the required Division 3 battery charger supplies ≥ 40 amps at ≥ 130 V for ≥ 4 hours.	24 months
SR 3.8.4.7	-------------------NOTES------------------- 1. The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7 provided the modified performance discharge test completely envelops the service test. 2. This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.8	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. -- Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months AND 12 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

LCO 3.8.5 DC electrical power subsystem(s) shall be OPERABLE to support the electrical power distribution subsystem(s) required by LCO 3.8.9, "Distribution Systems—Shutdown."

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

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel. AND A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-------------------NOTE------------------- The following SRs are not required to be performed: SR 3.8.4.7 and SR 3.8.4.8. -- For DC electrical power subsystems required to be OPERABLE the following SRs are applicable: SR 3.8.4.1, SR 3.8.4.2, SR 3.8.4.3, SR 3.8.4.4, SR 3.8.4.5, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the Division 1, 2, and 3 batteries shall be within limits.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category A or B limits.	A.1	Verify pilot cell(s) electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	<u>AND</u>		
	A.2	Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
	A.3	Restore battery cell parameters to Table 3.8.6-1 Category A and B limits.	31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with average electrolyte temperature of the representative cells < 65°F. OR One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category C limits.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days <u>AND</u> Once within 7 days after battery discharge < 107 V <u>AND</u> Once within 7 days after battery overcharge > 142 V
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ 65°F.	92 days

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameter Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[a]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[a]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[b][c]	≥ 1.200	≥ 1.195 AND Average of all connected cells > 1.205	Not more than 0.020 below average of all connected cells AND Average of all connected cells ≥ 1.195

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during and following equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters—Operating

LCO 3.8.7 The Division 1 and Division 2 emergency uninterruptible power supply (UPS) inverters shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One emergency UPS inverter inoperable.	A.1	---------NOTE-------- Enter applicable Conditions and Required Actions of LCO 3.8.8, "Distribution Systems—Operating" with any 120 VAC uninterruptible panel de-energized. -------------------- Restore emergency UPS inverters to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct emergency UPS inverter voltage, frequency, and alignment to 120 VAC uninterruptible panels.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems—Operating

LCO 3.8.8 The following AC and DC electrical power distribution subsystems shall be OPERABLE:

a. Division 1 and Division 2 AC electrical power distribution subsystems;

b. Division 1 and Division 2 120 VAC uninterruptible electrical power distribution subsystems;

c. Division 1 and Division 2 DC electrical power distribution subsystems; and

d. Division 3 AC and DC electrical power distribution subsystems.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or both Division 1 and 2 AC electrical power distribution subsystems inoperable.	A.1	Restore Division 1 and 2 AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO 3.8.8.a, b, or c

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or both Division 1 and 2 120 VAC uninterruptible electrical power distribution subsystems inoperable.	B.1	Restore Division 1 and 2 120 VAC uninterruptible electrical power distribution subsystem(s) to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO 3.8.8.a, b, or c
C. One or both Division 1 and 2 DC electrical power distribution subsystems inoperable.	C.1	Restore Division 1 and 2 DC electrical power distribution subsystem(s) to OPERABLE status.	2 hours AND 16 hours from discovery of failure to meet LCO 3.8.8.a, b, or c
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours
E. One or both Division 3 AC and DC electrical power distribution subsystems inoperable.	E.1	Declare High Pressure Core Spray System inoperable.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.8.1 Verify correct breaker alignments and power availability to required AC, DC, and 120 VAC uninterruptible electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems—Shutdown

LCO 3.8.9 The necessary portions of the Division 1, Division 2, and Division 3 AC and DC and the Division 1 and Division 2 120 VAC uninterruptible electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

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

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or 120 VAC uninterruptible electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel. AND A.2.4 Initiate actions to restore required AC, DC, and 120 VAC uninterruptible electrical power distribution subsystems to OPERABLE status. AND A.2.5 Declare associated required shutdown cooling subsystem(s) inoperable and not in operation.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and power availability to required AC, DC, and 120 VAC uninterruptible electrical power distribution subsystems.	7 days

3.9 REFUELING OPERATIONS

3.9.1 Refueling Equipment Interlocks

LCO 3.9.1 The refueling equipment interlocks associated with the reactor mode switch refuel position shall be OPERABLE.

APPLICABILITY: During in-vessel fuel movement with equipment associated with the interlocks when the reactor mode switch is in the refuel position.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required refueling equipment interlocks inoperable.	A.1 Suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s).	Immediately
	OR	
	A.2.1 Insert a control rod withdrawal block.	Immediately
	AND	
	A.2.2 Verify all control rods are fully inserted in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Perform CHANNEL FUNCTIONAL TEST on each of the following required refueling equipment interlock inputs: a. All-rods-in, b. Refueling platform position, c. Refueling platform fuel grapple, fuel-loaded, d. Refueling platform monorail hoist, fuel-loaded, e. Refueling platform frame-mounted hoist, fuel-loaded, and f. Service platform hoist, fuel-loaded.	7 days

3.9 REFUELING OPERATIONS

3.9.2 Refuel Position One-Rod-Out Interlock

LCO 3.9.2 The refuel position one-rod-out interlock shall be OPERABLE.

APPLICABILITY: MODE 5 with the reactor mode switch in the refuel position and any control rod withdrawn.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refuel position one-rod-out interlock inoperable.	A.1 Suspend control rod withdrawal.	Immediately
	<u>AND</u>	
	A.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Verify reactor mode switch locked in refuel position.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.9.2.2	------------------NOTE------------------ Not required to be performed until 1 hour after any control rod is withdrawn. -- Perform CHANNEL FUNCTIONAL TEST.	7 days

3.9 REFUELING OPERATIONS

3.9.3 Control Rod Position

LCO 3.9.3 All control rods shall be fully inserted.

APPLICABILITY: When loading fuel assemblies into the core.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more control rods not fully inserted.	A.1 Suspend loading fuel assemblies into the core.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.3.1 Verify all control rods are fully inserted.	12 hours

3.9 REFUELING OPERATIONS

3.9.4 Control Rod Position Indication

LCO 3.9.4 Each control rod "full-in" position indication channel shall be OPERABLE.

APPLICABILITY: MODE 5.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more control rod position indication channels inoperable.	A.1.1 Suspend in-vessel fuel movement.	Immediately
	<u>AND</u>	
	A.1.2 Suspend control rod withdrawal.	Immediately
	<u>AND</u>	
	A.1.3 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	<u>OR</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.1 Initiate action to fully insert the control rod associated with the inoperable position indicator.	Immediately
	AND	
	A.2.2 Initiate action to disarm the control rod drive associated with the fully inserted control rod.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each channel has no "full-in" indication on each control rod that is not "full-in."	Each time the control rod is withdrawn from the "full-in" position

3.9 REFUELING OPERATIONS

3.9.5 Control Rod OPERABILITY—Refueling

LCO 3.9.5 Each withdrawn control rod shall be OPERABLE.

APPLICABILITY: MODE 5.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more withdrawn control rods inoperable.	A.1	Initiate action to fully insert inoperable withdrawn control rods.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	-------------------NOTE------------------- Not required to be performed until 7 days after the control rod is withdrawn. -- Insert each withdrawn control rod at least one notch.	7 days
SR 3.9.5.2	Verify each withdrawn control rod scram accumulator pressure is ≥ 940 psig.	7 days

3.9 REFUELING OPERATIONS

3.9.6 Reactor Pressure Vessel (RPV) Water Level—Irradiated Fuel

LCO 3.9.6 RPV water level shall be ≥ 22 ft 3 inches above the top of the RPV flange.

APPLICABILITY: During movement of irradiated fuel assemblies within the RPV.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RPV water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within the RPV.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.6.1 Verify RPV water level is ≥ 22 ft 3 inches above the top of the RPV flange.	24 hours

3.9 REFUELING OPERATIONS

3.9.7 Reactor Pressure Vessel (RPV) Water Level—New Fuel or Control Rods

LCO 3.9.7 RPV water level shall be ≥ 22 ft 3 inches above the top of irradiated fuel assemblies seated within the RPV.

APPLICABILITY: During movement of new fuel assemblies or handling of control rods within the RPV when irradiated fuel assemblies are seated within the RPV.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RPV water level not within limit.	A.1 Suspend movement of new fuel assemblies and handling of control rods within the RPV.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify RPV water level is ≥ 22 ft 3 inches above the top of irradiated fuel assemblies seated within the RPV.	24 hours

3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR)—High Water Level

LCO 3.9.8 One RHR shutdown cooling subsystem shall be OPERABLE and in operation.

---NOTE---
The required RHR shutdown cooling subsystem may be not in operation for up to 2 hours per 8 hour period.

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level $\geq$ 22 ft 3 inches above the top of the RPV flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required RHR shutdown cooling subsystem inoperable.	A.1 Verify an alternate method of decay heat removal is available.	1 hour <u>AND</u> Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Suspend loading irradiated fuel assemblies into the RPV. <u>AND</u>	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Initiate action to restore secondary containment to OPERABLE status.	Immediately
	<u>AND</u>		
	B.3	Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
	<u>AND</u>		
	B.4	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	Immediately
C. No RHR shutdown cooling subsystem in operation.	C.1	Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>		
	C.2	Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.8.1	Verify one RHR shutdown cooling subsystem is operating.	12 hours

3.9 REFUELING OPERATIONS

3.9.9 Residual Heat Removal (RHR)—Low Water Level

LCO 3.9.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and one RHR shutdown cooling subsystem shall be in operation.

---------------------------NOTE---------------------------
The required operating shutdown cooling subsystem may be not in operation for up to 2 hours per 8 hour period.

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level < 22 ft 3 inches above the top of the RPV flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
-----------NOTE----------- Separate Condition entry is allowed for each inoperable RHR shutdown cooling subsystem. -------------------------- A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for the inoperable RHR shutdown cooling subsystem.	1 hour AND Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to restore secondary containment to OPERABLE status. AND	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
	<u>AND</u>		
	B.3	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	Immediately
C. No RHR shutdown cooling subsystem in operation.	C.1	Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>		
	C.2	Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.9.1	Verify one RHR shutdown cooling subsystem is operating.	12 hours

3.10 SPECIAL OPERATIONS

3.10.1 System Leakage and Hydrostatic Testing Operation

LCO 3.10.1

The average reactor coolant temperature specified in Table 1.1-1 for MODE 4 may be changed to "NA," and operation considered not to be in MODE 3; and the requirements of LCO 3.4.10, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown," may be suspended, to allow performance of a system leakage or hydrostatic test provided the following MODE 3 LCOs are met:

a. LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," Functions 1, 3, and 4 of Table 3.3.6.2-1;

b. LCO 3.3.8.2, "Reactor Protection System (RPS) Electric Power Monitoring—Logic";

c. LCO 3.6.4.1, "Secondary Containment";

d. LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)"; and

e. LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABILITY: MODE 4 with average reactor coolant temperature > 200°F.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each requirement of the LCO.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more of the above requirements not met.	A.1	---NOTE--- Required Actions to be in MODE 4 include reducing average reactor coolant temperature to ≤ 200°F. ---	
		Enter the applicable Condition of the affected LCO.	Immediately
	<u>OR</u>		
	A.2.1	Suspend activities that could increase the average reactor coolant temperature or pressure.	Immediately
	<u>AND</u>		
	A.2.2	Reduce average reactor coolant temperature to ≤ 200°F.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.1.1	Perform the applicable SRs for the required MODE 3 LCOs.	According to the applicable SRs

3.10 SPECIAL OPERATIONS

3.10.2 Reactor Mode Switch Interlock Testing

LCO 3.10.2 The reactor mode switch position specified in Table 1.1-1 for MODES 3, 4, and 5 may be changed to include the run, startup/hot standby, and refuel position, and operation considered not to be in MODE 1 or 2, to allow testing of instrumentation associated with the reactor mode switch interlock functions, provided:

a. All control rods remain fully inserted in core cells containing one or more fuel assemblies; and

b. No CORE ALTERATIONS are in progress.

APPLICABILITY: MODES 3 and 4 with the reactor mode switch in the run, startup/hot standby, or refuel position,
MODE 5 with the reactor mode switch in the run or startup/hot standby position.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the above requirements not met.	A.1 Suspend CORE ALTERATIONS except for control rod insertion.	Immediately
	<u>AND</u>	
	A.2 Fully insert all insertable control rods in core cells containing one or more fuel assemblies.	1 hour
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3.1 Place the reactor mode switch in the shutdown position. <u>OR</u> A.3.2 --------NOTE--------- Only applicable in MODE 5. ---------------------	1 hour
	Place the reactor mode switch in the refuel position.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.2.1	Verify all control rods are fully inserted in core cells containing one or more fuel assemblies.	12 hours
SR 3.10.2.2	Verify no CORE ALTERATIONS are in progress.	24 hours

3.10 SPECIAL OPERATIONS

3.10.3 Single Control Rod Withdrawal—Hot Shutdown

LCO 3.10.3 The reactor mode switch position specified in Table 1.1-1 for MODE 3 may be changed to include the refuel position, and operation considered not to be in MODE 2, to allow withdrawal of a single control rod, provided the following requirements are met:

a. LCO 3.9.2, "Refuel Position One-Rod-Out Interlock";

b. LCO 3.9.4, "Control Rod Position Indication";

c. All other control rods are fully inserted; and

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

LCO 3.3.8.3, "Reactor Protection System (RPS) Electric Power Monitoring—Scram Solenoids," MODE 5 requirements, and

LCO 3.9.5, "Control Rod OPERABILITY—Refueling,"

<u>OR</u>

2. All other control rods in a five by five array centered on the control rod being withdrawn are disarmed, at which time LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," MODE 3 requirements may be changed to allow the single control rod withdrawn to be assumed to be the highest worth control rod.

APPLICABILITY: MODE 3 with the reactor mode switch in the refuel position.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each requirement of the LCO.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more of the above requirements not met.	A.1	--------NOTES-------- 1. Required Actions to fully insert all insertable control rods include placing the reactor mode switch in the shutdown position. 2. Only applicable if the requirement not met is a required LCO. -------------------- Enter the applicable Condition of the affected LCO.	Immediately
	<u>OR</u>		
	A.2.1	Initiate action to fully insert all insertable control rods.	Immediately
	<u>AND</u>		
	A.2.2	Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.3.1	Perform the applicable SRs for the required LCOs.	According to the applicable SRs
SR 3.10.3.2	-------------------NOTE------------------- Not required to be met if SR 3.10.3.1 is satisfied for LCO 3.10.3.d.1 requirements. -- Verify all control rods, other than the control rod being withdrawn, in a five by five array centered on the control rod being withdrawn, are disarmed.	24 hours
SR 3.10.3.3	Verify all control rods, other than the control rod being withdrawn, are fully inserted.	24 hours

3.10 SPECIAL OPERATIONS

3.10.4 Single Control Rod Withdrawal—Cold Shutdown

LCO 3.10.4 The reactor mode switch position specified in Table 1.1-1 for MODE 4 may be changed to include the refuel position, and operation considered not to be in MODE 2, to allow withdrawal of a single control rod, and subsequent removal of the associated control rod drive (CRD) if desired, provided the following requirements are met:

a. All other control rods are fully inserted;

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

LCO 3.9.4, "Control Rod Position Indication,"

<u>OR</u>

2. A control rod withdrawal block is inserted; and

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

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

LCO 3.3.8.3, "Reactor Protection System (RPS) Electric Power Monitoring—Scram Solenoids," MODE 5 requirements, and

LCO 3.9.5, "Control Rod OPERABILITY—Refueling,"

<u>OR</u>

2. All other control rods in a five by five array centered on the control rod being withdrawn are disarmed, at which time LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," MODE 5 requirements, may be changed to allow the single control rod withdrawn to be assumed to be the highest worth control rod.

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

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each requirement of the LCO.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the above requirements not met with the affected control rod insertable.	A.1 --------NOTES-------- 1. Required Actions to fully insert all insertable control rods include placing the reactor mode switch in the shutdown position. 2. Only applicable if the requirement not met is a required LCO. -------------------- Enter the applicable Condition of the affected LCO.	Immediately
	<u>OR</u>	
	A.2.1 Initiate action to fully insert all insertable control rods.	Immediately
	<u>AND</u>	
	A.2.2 Place the reactor mode switch in the shutdown position.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more of the above requirements not met with the affected control rod not insertable.	B.1 Suspend withdrawal of the control rod and removal of associated CRD.	Immediately
	<u>AND</u>	
	B.2.1 Initiate action to fully insert all control rods.	Immediately
	<u>OR</u>	
	B.2.2 Initiate action to satisfy the requirements of this LCO.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.10.4.1	Perform the applicable SRs for the required LCOs.	According to applicable SRs
SR 3.10.4.2	-------------------NOTE------------------- Not required to be met if SR 3.10.4.1 is satisfied for LCO 3.10.4.c.1 requirements. --- Verify all control rods, other than the control rod being withdrawn, in a five by five array centered on the control rod being withdrawn, are disarmed.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.10.4.3	Verify all control rods, other than the control rod being withdrawn, are fully inserted.	24 hours
SR 3.10.4.4	-------------------NOTE------------------- Not required to be met if SR 3.10.4.1 is satisfied for LCO 3.10.4.b.1 requirements. -- Verify a control rod withdrawal block is inserted.	24 hours

3.10 SPECIAL OPERATIONS

3.10.5 Single Control Rod Drive (CRD) Removal—Refueling

LCO 3.10.5 The requirements of LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"; LCO 3.3.8.2, "Reactor Protection System (RPS) Electric Power Monitoring—Logic"; LCO 3.3.8.3, "Reactor Protection System (RPS) Electric Power Monitoring—Scram Solenoids"; LCO 3.9.1, "Refueling Equipment Interlocks"; LCO 3.9.2, "Refuel Position One-Rod-Out Interlock"; LCO 3.9.4, "Control Rod Position Indication"; and LCO 3.9.5, "Control Rod OPERABILITY—Refueling," may be suspended in MODE 5 to allow the removal of a single CRD associated with a control rod withdrawn from a core cell containing one or more fuel assemblies, provided the following requirements are met:

a. All other control rods are fully inserted;

b. All other control rods in a five by five array centered on the withdrawn control rod are disarmed;

c. A control rod withdrawal block is inserted, and LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," MODE 5 requirements may be changed to allow the single control rod withdrawn to be assumed to be the highest worth control rod; and

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: MODE 5 with LCO 3.9.5 not met.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the above requirements not met.	A.1 Suspend removal of the CRD mechanism. AND	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.1 Initiate action to fully insert all control rods. OR A.2.2 Initiate action to satisfy the requirements of this LCO.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.5.1	Verify all controls rods, other than the control rod withdrawn for the removal of the associated CRD, are fully inserted.	24 hours
SR 3.10.5.2	Verify all control rods, other than the control rod withdrawn for the removal of the associated CRD, in a five by five array centered on the control rod withdrawn for the removal of the associated CRD, are disarmed.	24 hours
SR 3.10.5.3	Verify a control rod withdrawal block is inserted.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.10.5.4	Perform SR 3.1.1.1.	According to SR 3.1.1.1
SR 3.10.5.5	Verify no other CORE ALTERATIONS are in progress.	24 hours

3.10 SPECIAL OPERATIONS

3.10.6 Multiple Control Rod Withdrawal—Refueling

LCO 3.10.6 The requirements of LCO 3.9.3, "Control Rod Position"; LCO 3.9.4, "Control Rod Position Indication"; and LCO 3.9.5, "Control Rod OPERABILITY—Refueling," may be suspended, and the "full-in" position indicators may be bypassed for any number of control rods in MODE 5, to allow withdrawal of these control rods, removal of associated control rod drives (CRDs), or both, provided the following requirements are met:

a. The four fuel assemblies are removed from the core cells associated with each control rod or CRD to be removed;

b. All other control rods in core cells containing one or more fuel assemblies are fully inserted; and

c. Fuel assemblies shall only be loaded in compliance with an approved spiral reload sequence.

APPLICABILITY: MODE 5 with LCO 3.9.3, LCO 3.9.4, or LCO 3.9.5 not met.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the above requirements not met.	A.1 Suspend withdrawal of control rods and removal of associated CRDs.	Immediately
	<u>AND</u>	
	A.2 Suspend loading fuel assemblies.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3.1 Initiate action to fully insert all control rods in core cells containing one or more fuel assemblies.	Immediately
	<u>OR</u>	
	A.3.2 Initiate action to satisfy the requirements of this LCO.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.10.6.1	Verify the four fuel assemblies are removed from core cells associated with each control rod or CRD removed.	24 hours
SR 3.10.6.2	Verify all other control rods in core cells containing one or more fuel assemblies are fully inserted.	24 hours
SR 3.10.6.3	-------------------NOTE------------------- Only required to be met during fuel loading. -- Verify fuel assemblies being loaded are in compliance with an approved spiral reload sequence.	24 hours

3.10 SPECIAL OPERATIONS

3.10.7 Control Rod Testing—Operating

LCO 3.10.7 The requirements of LCO 3.1.6, "Rod Pattern Control," may be suspended, to allow performance of SDM demonstrations, control rod scram time testing, and control rod friction testing provided:

a. The banked position withdrawal sequence requirements of SR 3.3.2.1.8 are changed to require the control rod sequence to conform to the specified test sequence.

OR

b. The RWM is bypassed; the requirements of LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 2 are suspended; and conformance to the approved control rod sequence for the specified test is verified by a second licensed operator or other qualified member of the technical staff.

APPLICABILITY: MODES 1 and 2 with LCO 3.1.6 not met.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Suspend performance of the test and exception to LCO 3.1.6.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.10.7.1	-------------------NOTE------------------- Not required to be met if SR 3.10.7.2 satisfied. -- Verify movement of control rods is in compliance with the approved control rod sequence for the specified test by a second licensed operator or other qualified member of the technical staff.	During control rod movement
SR 3.10.7.2	-------------------NOTE------------------- Not required to be met if SR 3.10.7.1 satisfied. -- Verify control rod sequence input to the RWM is in conformance with the approved control rod sequence for the specified test.	Prior to control rod movement

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test—Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

a. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," MODE 2 requirements for Functions 2.a, 2.d, and 2.e of Table 3.3.1.1-1;

b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with the banked position withdrawal sequence requirements of SR 3.3.2.1.8 changed to require the control rod sequence to conform to the SDM test sequence,

   OR

   2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

c. Each withdrawn control rod shall be coupled to the associated CRD;

d. All control rod withdrawals during out of sequence control rod moves shall be made in single notch withdrawal mode;

e. No other CORE ALTERATIONS are in progress; and

f. CRD charging water header pressure ≥ 940 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
-----------NOTE----------- Separate Condition entry is allowed for each control rod. -------------------------- A. One or more control rods not coupled to its associated CRD.	------------NOTE------------- Rod Worth Minimizer may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ----------------------------- A.1 Fully insert inoperable control rod. <u>AND</u> A.2 Disarm the associated CRD.	 3 hours 4 hours
B. One or more of the above requirements not met for reasons other than Condition A.	B.1 Place the reactor mode switch in the shutdown or refuel position.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.10.8.1 Perform the MODE 2 applicable SRs for LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e of Table 3.3.1.1-1.	According to the applicable SRs

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.10.8.2	------------------NOTE------------------ Not required to be met if SR 3.10.8.3 satisfied. -- Perform the MODE 2 applicable SRs for LCO 3.3.2.1, Function 2 of Table 3.3.2.1-1.	According to the applicable SRs
SR 3.10.8.3	------------------NOTE------------------ Not required to be met if SR 3.10.8.2 satisfied. -- Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff.	During control rod movement
SR 3.10.8.4	Verify no other CORE ALTERATIONS are in progress.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.10.8.5	Verify each withdrawn control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position <u>AND</u> Prior to satisfying LCO 3.10.8.c requirement after work on control rod or CRD System that could affect coupling
SR 3.10.8.6	Verify CRD charging water header pressure ≥ 940 psig.	7 days

4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 Site and Exclusion Area Boundaries

The site area boundary and the land portion of the exclusion area boundary are as shown in Figure 4.1-1. The lake portion of the exclusion area boundary is the area of Lake Ontario within a 2 mile radius of the Nine Mile Point Unit 2 reactor centerline.

4.1.2 Low Population Zone

The low population zone is all the land within a circle with its center at the Nine Mile Point Unit 1 stack and a radius of four miles.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material, and water rods. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead fuel assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide and hafnium metal as approved by the NRC.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the USAR; and

b. A nominal 6.18 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1 of the USAR;

b. $k_{eff} \leq 0.98$ with all but one of the non-combustible storage vaults covers in place when optimum moderation (foam, spray, fogging, or small droplets) is assumed; and

c. A nominal 7.00 inch center to center distance between fuel assemblies placed within a storage rack and a nominal 12.25 inch center to center distance between fuel assemblies in adjacent racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 329 ft 7 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4049 fuel assemblies.

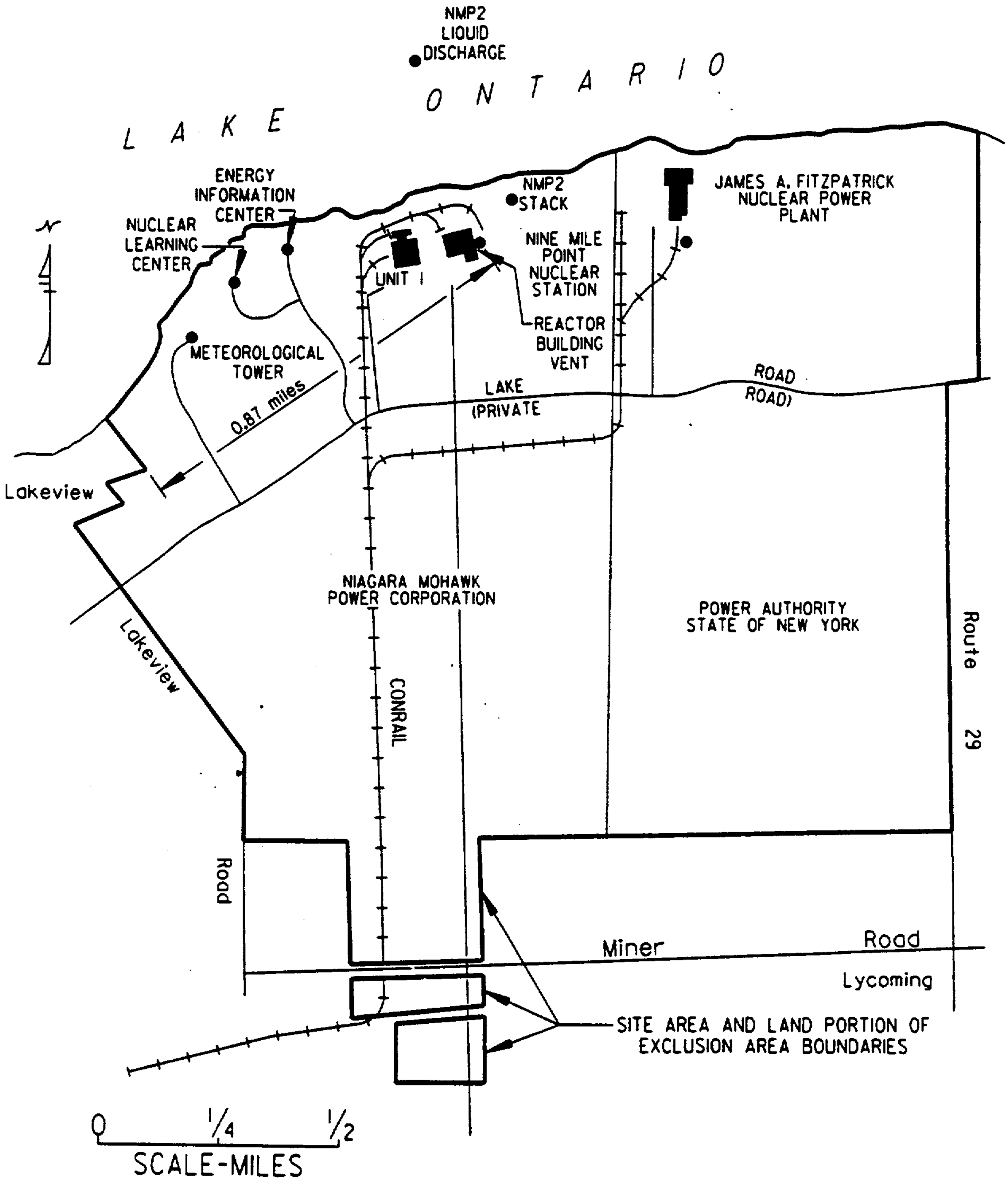

Figure 4.1-1 (Page 1 of 1)
Site Area and Land Portion of Exclusion Area Boundaries

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

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

The plant manager or a designee shall approve, prior to implementation, each proposed test and experiment not addressed in the USAR or Technical Specifications, and each modification to systems or equipment that affect nuclear safety.

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

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. The organizational chart and the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the USAR. The functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions shall be documented in procedures.

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff, carry out radiation protection, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

(continued)

5.2 Organization (continued)

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. At least two non-licensed operators shall be assigned when the unit is in MODE 1, 2, or 3; and at least one non-licensed operator shall be assigned when the unit is in MODE 4 or 5. In addition, if the process computer is out of service for greater than 8 hours, at least three non-licensed operators shall be assigned when the unit is in MODE 1, 2, 3, 4, or 5.

b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, or 3, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

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

d. An individual qualified to implement radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence of on-duty personnel, provided immediate action is taken to fill the required position.

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, key radiation protection personnel, auxiliary operators, and key maintenance personnel).

   Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an 8 to 12 hour day, nominal 40 hour week, while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

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

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time; and

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized in advance by the plant manager or a designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by a specified corporate officer or a designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

f. The operations supervisors shall hold an SRO license.

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

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions, except for the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for radioactive effluent and radiological environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The 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 and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   (a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   (b) A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after the approval of the plant manager or a designee; and

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of, or concurrent with, the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made.

(continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include the Low Pressure Core Spray, High Pressure Core Spray, Residual Heat Removal, Reactor Core Isolation Cooling, hydrogen recombiner, process sampling, containment monitoring and Standby Gas Treatment. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at 24 month intervals.

The provisions of SR 3.0.2 are applicable to the 24 month Frequency for performing integrated system leak test activities.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

(continued)

5.5 Programs and Manuals (continued)

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

1. For noble gases: a dose rate ≤ 500 mrems/yr to the whole body and a dose rate ≤ 3000 mrems/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days: a dose rate ≤ 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190; and

k. Limitations on venting and purging of the primary containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequencies.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Table 3.9B-1 Note 5, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

(continued)

5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

a. Testing Frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda are as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems.

Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation.

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

Tests described in Specification 5.5.7.c shall be performed once per 24 months; after 720 hours of system operation; after any structural maintenance on the charcoal adsorber bank housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation.

Tests described in Specifications 5.5.7.d and 5.5.7.e shall be performed once per 24 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
Standby Gas Treatment (SGT) System	3600 to 4400
Control Room Envelope Filtration (CREF) System	2025 to 2475

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
SGT System	3600 to 4400
CREF System	2025 to 2475

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1979 (Method B) at a relative humidity greater than or equal to the value specified below:

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Penetration	RH
SGT System	0.175	95
CREF System	0.175	95

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	< 5.5	3600 to 4400
CREF System	< 5.5	2025 to 2475

e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below, adjusted to degraded voltage conditions, when tested in accordance with ANSI N510-1980:

ESF Ventilation System	Wattage (kW)
SGT System	14.0 to 17.1
CREF System	≥ 7.95

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen in the Main Condenser Offgas Treatment System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in all outside temporary liquid

(continued)

5.5 Programs and Manuals

5.5.8 Explosive Gas and Storage Tank Radioactive Monitoring Program (continued)

radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is ≤ 10 Ci, excluding tritium and dissolved or entrained noble gases.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program shall establish the required testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity, a specific gravity, or an absolute specific gravity within limits,

   2. A flash point and kinematic viscosity within limits for ASTM fuel oil,

   3. A clear and bright appearance;

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in 5.5.9.a above, are within limits for ASTM fuel oil; and

c. Total particulate concentration of the fuel oil in the storage tanks is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.10 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the USAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

d. Proposed changes that meet the criteria of 5.5.10.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.11 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

   1. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

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

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

5.5.12 10 CFR 50 Appendix J Testing Program Plan

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B with the exemptions stated in

(continued)

5.5 Programs and Manuals

5.5.12 10 CFR 50 Appendix J Testing Program Plan (continued)

Section 2.D(ii) of the Operating License. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled, "Performance-Based Containment Leak-Test Program," dated September 1995 with the following exceptions:

1. The measured leakage of main steam isolation valves (MSIVs) is excluded from the combined leakage rate of 0.6 L_a, and as-found testing is not required to be performed on the MSIVs.

2. Primary containment air lock door seals are tested prior to re-establishing primary containment OPERABILITY when something has been done that would bring into question the validity of the previous air lock door seal test.

b. The peak calculated containment internal pressure (P_a) for the design basis loss of coolant accident is 39.75 psig.

c. The maximum allowable primary containment leakage rate (L_a) at P_a shall be 1.1% of primary containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Primary Containment leakage rate acceptance criterion is < 1.0 L_a. The combined leakage rate for Type B and C tests on a minimum pathway basis, except for main steam line isolation valves and Primary Containment isolation valves which are hydrostatically tested, is < 0.6 L_a.

   During the first unit startup following testing in accordance with this program, the as-left combined leakage rate acceptance criteria are < 0.6 L_a for the Type B and C tests on a maximum pathway basis, except for main steam line isolation valves and Primary Containment isolation valves which are hydrostatically tested, and ≤ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

   (a) Overall air lock leakage rate is ≤ 0.05 L_a when tested at greater than or equal to P_a; and

(continued)

5.5 Programs and Manuals

5.5.12 10 CFR 50 Appendix J Testing Program Plan (continued)

(b) For each door, leakage rate is ≤ 5 scfh when the gap between the door seals is pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the 10 CFR 50 Appendix J Testing Program Plan.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

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

5.6.2 Annual Radiological Environmental Operating Report

-----------------------------NOTE-----------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

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

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. The APLHGR for Specification 3.2.1.

2. The MCPR for Specification 3.2.2.

3. The LHGR for Specification 3.2.3.

4. Control Rod Block Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel," U.S. Supplement, (NRC approved version specified in the COLR).

2. NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," (NRC approved version specified in the COLR).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20.

5.7.1 <u>High Radiation Areas with Dose Rates not Exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation)</u>

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess at least one of the following:

1. A radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device").

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached ("alarming dosimeter"), with an appropriate alarm setpoint.

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area.

(continued)

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates not Exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation) (continued)

4. A self-reading dosimeter and,

   (a) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual at the work site, qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel radiation exposure within the area, or

   (b) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door, gate, or guard that prevents unauthorized entry, and in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the Station Shift Supervisor - Nuclear or a designee, or the radiation protection manager or a designee; and

   2. Doors and gates shall remain locked or guarded except during periods of personnel entry or exit.

(continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work areas(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess at least one of the following:

1. An alarming dosimeter with an appropriate alarm setpoint.

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area.

3. A self-reading dosimeter and,

(a) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel exposure within the area, or

(continued)

5.7 High Radiation Area

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)</u> (continued)

(b) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. A radiation monitoring and indicating device in those cases where the options of Specifications 5.7.2.d.2 and 5.7.2.d.3, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp

(continued)

BASES

BACKGROUND (continued)

reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR Safety Limit.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28×10^3 lb/hr. Full scale ATLAS test

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in References 3 and 4. Reference 3 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and Reference 4 also provides the nominal values of the parameters used in the MCPR SL statistical analysis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	2.1.1.3 Reactor Vessel Water Level During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.
SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	2.2 Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 5). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. GE Service Information Letter No. 516, Supplement 2, "Core Flow Indication in the Low-Flow Region," January 19, 1996.
3. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).
4. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2 (revision specified in the COLR).
5. 10 CFR 100.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

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

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, the number of protective barriers designed to prevent radioactive releases from exceeding the limits would be reduced.

APPLICABLE SAFETY ANALYSES

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, including Addenda through the winter of 1972 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to ASME Code, Section III, 1977 Edition, including Addenda through the summer of 1977 (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 1250 psig for suction piping up to the reactor recirculation pump, 1650 psig for discharge piping up to and including the discharge blocking valve, and 1550 psig for the piping after the discharge blocking valve. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.
SAFETY LIMITS	The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 110% of design pressures of 1250 psig for suction piping up to the reactor recirculation pump, 1650 psig for discharge piping up to and including the discharge blocking valve, and 1550 psig for the piping after the discharge blocking valve. The most limiting of these allowances is the 110% of the reactor vessel and the suction piping up to the reactor recirculation pump design pressures; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.
APPLICABILITY	SL 2.1.2 applies in all MODES.
SAFETY LIMIT VIOLATIONS	<u>2.2</u> Exceeding the RCS pressure SL may cause RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control

(continued)

BASES

SAFETY LIMIT VIOLATIONS	2.2 (continued) rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 14 and GDC 15. 2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000. 3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWA-5000. 4. 10 CFR 100. 5. ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, Addenda, winter of 1972. 6. ASME, Boiler and Pressure Vessel Code, Section III, 1977 Edition, Addenda, summer of 1977.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering

(continued)

BASES

LCO 3.0.2 (continued)

ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Condition no longer exists. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.11, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/divisions of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

(continued)

BASES (continued)

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

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BASES

LCO 3.0.3 (continued)

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 4 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 4, or other applicable MODE, is not reduced. For example, if MODE 2 is reached in 2 hours, then the time allowed for reaching MODE 3 is the next 11 hours, because the total time for reaching MODE 3 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, and 3, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 4 and 5 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do nct apply in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.6, "Spent Fuel Storage Pool Water Level." LCO 3.7.6 has an Applicability of "During movement of irradiated fuel

(continued)

BASES

LCO 3.0.3 (continued)

assemblies in the spent fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.6 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.6 of "Suspend movement of irradiated fuel assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

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BASES

LCO 3.0.4 (continued)	The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification. Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, either in compliance with LCO 3.0.4, or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO. LCO 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of LCO 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.
LCO 3.0.5	LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

(continued)

BASES

LCO 3.0.5 (continued)

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

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

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

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system's LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the plant is maintained in a safe condition are specified in the support systems' LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability.

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BASES

LCO 3.0.6 (continued)

However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCO's Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.11, "Safety Function Determination Program" (SFDP), ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross division checks to identify a loss of safety function for those support systems that support safety systems are required. The cross division check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operation is being restricted in accordance with the

(continued)

BASES

LCO 3.0.6 (continued)

ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross division inoperabilities. This explicit cross division verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABLE—OPERABILITY).

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.10 allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Special Operations LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Special Operations

(continued)

BASES

LCO 3.0.7 (continued)	LCOs is optional. A special operation may be performed either under the provisions of the appropriate Special Operations LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Special Operations LCO, the requirements of the Special Operations LCO shall be followed. When a Special Operations LCO requires another LCO to be met, only the requirements of the LCO statement are required to be met regardless of that LCO's Applicability (i.e., should the requirements of this other LCO not be met, the ACTIONS of the Special Operations LCO apply, not the ACTIONS of the other LCO). However, there are instances where the Special Operations LCO's ACTIONS may direct the other LCO's ACTIONS be met. The Surveillances of the other LCO are not required to be met, unless specified in the Special Operations LCO. If conditions exist such that the Applicability of any other LCO is met, all the other LCO's requirements (ACTIONS and SRs) are required to be met concurrent with the requirements of the Special Operations LCO.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Operations LCO are only applicable when the Special Operations LCO is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

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BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed. Some examples of this process are:

a. Control rod drive maintenance during refueling that requires scram testing at ≥ 800 psig. However, if other appropriate testing is satisfactorily completed and the scram time testing of SR 3.1.4.3 is satisfied, the control rod can be considered OPERABLE. This allows startup to proceed to reach 800 psig to perform other necessary testing.

b. Reactor Core Isolation Cooling (RCIC) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with RCIC considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

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

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

(continued)

BASES

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the SR includes a Note in the Frequency stating "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

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

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have

(continued)

BASES

SR 3.0.3 (continued)

been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable then is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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BASES (continued)

SR 3.0.4

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

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

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1 which states that Surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency, on equipment that is inoperable, does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows

(continued)

BASES

SR 3.0.4 (continued)

performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of SR 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

These requirements are satisfied by the control rods, as described in GDC 26 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

The control rod drop accident (CRDA) analysis (Refs. 2 and 3) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also, SDM is assumed as an initial condition for the control rod removal error during a refueling accident (Ref. 4). The analysis of this reactivity insertion event assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal—Refueling.") The analysis assumes this condition is acceptable since the core will be shut down with the highest worth control rod withdrawn, if adequate SDM has been demonstrated.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential CRDAs involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage. SDM satisfies Criterion 2 of Reference 5.
LCO	The specified SDM limit accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin is included to account for uncertainties in the calculation. To ensure adequate SDM, a design margin is included to account for uncertainties in the design calculations (Ref. 6).
APPLICABILITY	In MODES 1 and 2, SDM must be provided because subcriticality with the highest worth control rod withdrawn is assumed in the CRDA analysis (Ref. 3). In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in MODE 5 to prevent an inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies (Ref. 4).
ACTIONS	<u>A.1</u> With SDM not within the limits of the LCO in MODE 1 or 2, SDM must be restored within 6 hours. Failure to meet the specified SDM may be caused by a control rod that cannot be inserted. The 6 hour Completion time is acceptable, considering that the reactor can still be shut down, assuming no additional failures of control rods to insert, and the low probability of an event occurring during this interval.

(continued)

BASES

ACTIONS (continued)

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 within 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

With SDM not within limits in MODE 3, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core.

D.1, D.2, D.3, and D.4

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Actions must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one Standby Gas Treatment (SGT) subsystem is OPERABLE; and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate

(continued)

BASES

ACTIONS

D.1, D.2, D.3, and D.4 (continued)

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

E.1, E.2, E.3, E.4, and E.5

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM, e.g., insertion of fuel in the core or the withdrawal of control rods. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one SGT subsystem is OPERABLE; and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). This may be performed as an administrative check, by examining logs or

(continued)

BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Adequate SDM must be verified to ensure the reactor can be made subcritical from any initial operating condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement or shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (i.e., BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 7). For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% $\Delta k/k$) must be added to the SDM limit of 0.28% $\Delta k/k$ to account for uncertainties in the calculation.

The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods. This testing would therefore require

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing—Operating").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and appropriate verification.

During MODES 3 and 4, analytical calculation of SDM may be used to assure the requirements of SR 3.1.1.1 are met. During MODE 5, adequate SDM is also required to ensure the reactor does not reach criticality during control rod withdrawals. An evaluation of each in vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload or reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. USAR, Section 15.4.9.
3. NEDO-21231, "Banked Position Withdrawal Sequence," Section 4.1, January 1977.
4. USAR, Section 15.4.1.1.
5. 10 CFR 50.36(c)(2)(ii).
6. USAR, Section 4.3.2.4.1.
7. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Reactivity Anomalies is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable

(continued)

BASES

BACKGROUND (continued)

absorbers (e.g., gadolinia), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel.

The predicted core reactivity, as represented by control rod density, is calculated by a 3D core simulator code as a function of cycle exposure. Rod density shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% rod density. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined for control rod densities for actual plant conditions and is then compared to the predicted value for the cycle exposure.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted rod density for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod density may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured rod density from the predicted rod density that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred.

Reactivity Anomalies satisfies Criterion 2 of Reference 3.

(continued)

BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the difference between the monitored rod density and the predicted rod density of 1% $\Delta k/k$ has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, all control rods are fully inserted, and, therefore, the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, Reactivity Anomalies is not required during these conditions.

ACTIONS

A.1

Should an anomaly develop between measured and predicted core reactivity, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core conditions to determine their consistency with input to design calculations. Measured core and process parameters

(continued)

BASES

ACTIONS

A.1 (continued)

are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, 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. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the monitored and predicted rod density is within the limits of the LCO provides further assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The core monitoring system calculates the rod density for the reactor conditions obtained from plant instrumentation. A comparison of the monitored rod density to the predicted rod density at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted rod density values can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (at equilibrium xenon with no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. USAR, Chapter 15.

3. 10 CFR 50.36(c)(2)(ii).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the Control Rod Drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure that under conditions of normal operation, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, GDC 27, GDC 28, and GDC 29, (Ref. 1).

The CRD System consists of 185 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," LCO 3.1.5, "Control Rod Scram Accumulators," and LCO 3.1.6, "Rod Pattern Control," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, 4, 5, and 6.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, 4, 5, and 6. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical, and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

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BASES

APPLICABLE SAFETY ANALYSES (continued)	The capability of inserting the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function. The control rods also protect the fuel from damage that could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events. The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System. Control rod OPERABILITY satisfies Criterion 3 of Reference 7.
LCO	OPERABILITY of an individual control rod is based on a combination of factors, primarily the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times and therefore an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to

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BASES

LCO (continued)

satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

OPERABILITY requirements for control rods also includes correct assembly of the CRD housing supports. These supports prevent any significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel. Correct assembly of the CRD housing supports is ensured by satisfying the two criteria specified in References 8, 9, and 10.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

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

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note that allows the Rod Worth Minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram activity rate assumptions may not be met if the stuck

(continued)

BASES

ACTIONS

A.1, A.2, A.3, and A.4 (continued)

control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The stuck control rod separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) if the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods elsewhere in the core adjacent to one another. The description of "slow" control rods is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed within 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable amount of time to perform the Required Action in an orderly manner. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram and normal insert and withdraw pressure prevents damage to the CRDM or reactor internals. The control rod isolation method should also ensure cooling water to the CRD is maintained.

Monitoring of the insertion capability for each withdrawn control rod must also be performed within 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does nct exist. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The Required Action A.3 Completion Time only begins upon discovery of Condition A concurrent with THERMAL POWER greater than the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The allowed Completion Time provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests.

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BASES

ACTIONS

A.1, A.2, A.3, and A.4 (continued)

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion an additional control rod would have to be assumed to have failed to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions (Ref. 11).

B.1

With two or more withdrawn control rods stuck, the plant must be brought to MODE 3 within 12 hours. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically, the control rods can be disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note that allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

D.1 and D.2

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ 10% RTP, the generic banked position withdrawal sequence (BPWS) analysis (Ref. 8) requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal (i.e., all other control rods in a five-by-five array centered on the inoperable control rod are OPERABLE). Therefore, if two or more inoperable control rods are not in compliance with BPWS and not separated by at least two OPERABLE control rods in all directions, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. A Note has been added to the Condition to clarify that the Condition is not applicable when > 10% RTP since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.

(continued)

BASES

ACTIONS (continued)

E.1

If any Required Action and associated Completion Time of Condition A, C, or D are not met or nine or more inoperable control rods exist, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above 10% RTP (i.e., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

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

SR 3.1.3.2 and SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are not required when THERMAL POWER is less than or equal to the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 and SR 3.1.3.3 (continued)

actual LPSP of the RWM since the notch insertions may not be compatible with the requirements of the banked position withdrawal sequence (BPWS) (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The 7 day Frequency of SR 3.1.3.2 is based on operating experience related to the changes in CRD performance and the ease of performing notch testing for fully withdrawn control rods. Partially withdrawn control rods are tested at a 31 day Frequency, based on the potential power reduction required to allow the control rod movement, and considering the large testing sample of SR 3.1.3.2. Furthermore, the 31 day Frequency takes into account operating experience related to changes in CRD performance. At any time, if a control rod is immovable, a determination of that control rod's trippability (OPERABILITY) must be made and appropriate action taken.

These SRs are modified by Notes that allow 7 days and 31 days respectively, after withdrawal of the control rod and increasing power to above the LPSP, to perform the Surveillance. This acknowledges that the control rod must be first withdrawn and THERMAL POWER must be increased to above the LPSP before performance of the Surveillance, and therefore, the Notes avoid potential conflicts with SR 3.0.3 and SR 3.0.4.

SR 3.1.3.4

Verifying the scram time for each control rod to notch position 05 is ≤ 7 seconds provides reasonable assurance that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function. This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and the functional testing of SDV vent and drain valves in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlap this Surveillance to provide complete testing of the assumed safety function. The associated Frequencies are acceptable, considering the more frequent testing performed to demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.5

Coupling verification is performed to ensure the control rod is connected to the CRDM and will perform its intended function when necessary. The Surveillance requires verifying that a control rod does not go to the withdrawn overtravel position when it is fully withdrawn. The overtravel position feature provides a positive check on the coupling integrity, since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed anytime a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 27, GDC 28, and GDC 29.
2. USAR, Section 4.3.2.5.1.
3. USAR, Section 4.6.1.1.2.
4. USAR, Section 5.2.2.2.3.
5. USAR, Section 15.4.1.
6. USAR, Section 15.4.9.
7. 10 CFR 50.36(c)(2)(ii).
8. USAR, Section 4.6.1.2.
9. USAR, Section 4.6.2.3.3.
10. USAR, Section 4.6.3.2.
11. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

BASES

BACKGROUND

The scram function of the Control Rod Drive (CRD) System controls reactivity changes during abnormal operational occurrences to ensure that specified acceptable fuel design limits are not exceeded (Ref. 1). The control rods are scrammed by positive means, using hydraulic pressure exerted on the CRD piston.

When a scram signal is initiated, control air is vented from the scram valves, allowing them to open by spring action. Opening the exhaust valves reduces the pressure above the main drive piston to atmospheric pressure, and opening the inlet valve applies the accumulator or reactor pressure to the bottom of the piston. Since the notches in the index tube are tapered on the lower edge, the collet fingers are forced open by cam action, allowing the index tube to move upward without restriction because of the high differential pressure across the piston. As the drive moves upward and accumulator pressure drops below the reactor pressure, a ball check valve opens, letting the reactor pressure complete the scram action. If the reactor pressure is low, such as during startup, the accumulator will fully insert the control rod within the required time without assistance from reactor pressure.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 2, 3, 4, 5, and 6. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scramming slower than the average time, with several control rods scramming faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's scram time ensures the scram reactivity assumed in the DBA and transient analyses can be met.

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"),

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," which ensure that no fuel damage will occur if these limits are not exceeded. Above 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident (Ref. 6) and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis (Ref. 4), the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

Control rod scram times satisfy Criterion 3 of Reference 7.

LCO

The scram times specified in Table 3.1.4-1 are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 8). To account for single failure and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin to allow up to 7.0% of the control rods (e.g., 185 x 7.0% ≈ 13) to have scram times that exceed the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times.

To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent (face or diagonal) locations.

(continued)

BASES

LCO (continued)

Table 3.1.4-1 is modified by two Notes, which state control rods with scram times not within the limits of the Table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods may be conservatively declared inoperable and not accounted for as "slow" control rods.

APPLICABILITY

In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

ACTIONS

A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analyses. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated (i.e., charging valve closed), the influence of the CRD pump head does not affect the single control rod scram times. During a full

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 5 and 6.

Maximum scram insertion times occur at a reactor pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure scram time testing is performed within a reasonable time following a shutdown ≥ 120 days, control rods are required to be tested before exceeding 40% RTP. This Frequency is acceptable, considering the additional Surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by fuel movement within the associated core cell and by work on control rods or the CRD System.

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. The sample remains representative if no more than 20% of the control rods in the sample tested are determined to be "slow." If more than 20% of the sample is declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 20% criterion (i.e., 20% of the entire sample size) is satisfied, or until the total number of "slow" control rods (throughout the core, from all Surveillances) exceeds the LCO limit. For

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.2 (continued)

planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data were previously tested in a sample. The 120 day Frequency is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle. This Frequency is also reasonable, based on the additional Surveillances done on the CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators."

SR 3.1.4.3

When work that could affect the scram insertion time is performed on a control rod or the CRD System, testing must be done to demonstrate that each affected control rod retains adequate scram performance over the range of applicable reactor pressures from zero to the maximum permissible pressure. The scram testing must be performed once before declaring the control rod OPERABLE. The required scram time testing must demonstrate that the affected control rod is still within acceptable limits. The scram time limits for reactor pressures < 800 psig are found in the Technical Requirements Manual (Ref. 9) and are established based on a high probability of meeting the acceptance criteria at reactor pressures ≥ 800 psig. Limits for reactor pressures ≥ 800 psig are found in Table 3.1.4-1. If testing demonstrates the affected control rod does not meet these limits, but is within 7-second limit of Table 3.1.4-1, Note 2, the control rod can be declared OPERABLE and "slow."

Specific examples of work that could affect the scram times include (but are not limited to) the following: removal of any CRD for maintenance or modification; replacement of a control rod; and maintenance or modification of a scram solenoid pilot valve, scram valve, accumulator isolation valve, or check valves in the piping required for scram.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

The Frequency of once prior to declaring the affected control rod OPERABLE is acceptable because of the capability of testing the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

SR 3.1.4.4

When work that could affect the scram insertion time is performed on a control rod or CRD System, or when fuel movement within the reactor pressure vessel occurs, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure ≥ 800 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 will be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and a high pressure test may be required. This testing ensures that the control rod scram performance is acceptable for operating reactor pressure conditions prior to withdrawing the control rod for continued operation. Alternatively, a test during hydrostatic pressure testing could also satisfy both criteria. When fuel movement within the reactor pressure vessel occurs, only those control rods associated with the core cells affected by the fuel movement are required to be scram time tested. During a routine refueling outage, it is expected that all control rods will be affected.

The Frequency of once prior to exceeding 40% RTP is acceptable because of the capability of testing the control rod at the different conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. USAR, Section 4.3.2.5.
3. USAR, Section 4.6.1.1.2.
4. USAR, Section 5.2.2.2.3.
5. USAR, Section 15.4.1.

(continued)

BASES

REFERENCES (continued)

6. USAR, Section 15.4.9.

7. 10 CFR 50.36(c)(2)(ii).

8. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specification," BWROG-8754, September 17, 1987.

9. Technical Requirements Manual.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND

The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 2, 3, and 4. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the DBA and transient analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod.

The scram function of the CRD System, and, therefore, the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). Also, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control").

Control rod scram accumulators satisfy Criterion 3 of Reference 5.

(continued)

BASES (continued)

LCO

The OPERABILITY of the control rod scram accumulators is required to ensure that adequate scram insertion capability exists when needed over the entire range of reactor pressures. The OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure.

APPLICABILITY

In MODES 1 and 2, the scram function is required for mitigation of DBAs and transients and, therefore, the scram accumulators must be OPERABLE to support the scram function. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram accumulator OPERABILITY under these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod scram accumulator. This is acceptable since the Required Actions for each Condition provide appropriate compensatory action for each inoperable accumulator. Complying with the Required Actions may allow for continued operation and subsequent inoperable accumulators governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

With one control rod scram accumulator inoperable and the reactor steam dome pressure ≥ 900 psig, the control rod may be declared "slow," since the control rod will still scram at the reactor operating pressure but may not satisfy the required scram times in Table 3.1.4-1. Required Action A.1 is modified by a Note, which clarifies that declaring the control rod "slow" is only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time surveillance. Otherwise, the control rod may already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action A.2) and LCO 3.1.3 entered. This would result in requiring the affected

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 8 hours is considered reasonable, based on the large number of control rods available to provide the scram function and the ability of the affected control rod to scram only with reactor pressure at high reactor pressures.

B.1, B.2.1, and B.2.2

With two or more control rod scram accumulators inoperable and reactor steam dome pressure ≥ 900 psig, adequate pressure must be supplied to the charging water header. With inadequate charging water pressure, all of the accumulators could become inoperable, resulting in a potentially severe degradation of the scram performance. Therefore, within 20 minutes from discovery of charging water header pressure < 940 psig concurrent with Condition B, adequate charging water header pressure must be restored. The allowed Completion Time of 20 minutes is considered a reasonable time to place a CRD pump into service to restore the charging header pressure, if required. This Completion Time also recognizes the ability of the reactor pressure alone to fully insert all control rods.

The control rod may be declared "slow," since the control rod will still scram using only reactor pressure, but may not satisfy the times in Table 3.1.4-1. Required Action B.2.1 is modified by a Note indicating that declaring the control rod "slow" is only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time surveillance. Otherwise, the control rod may already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action B.2.2) and LCO 3.1.3 entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

(continued)

BASES

ACTIONS

B.1, B.2.1, and B.2.2 (continued)

The allowed Completion Time of 1 hour is considered reasonable, based on the ability of only the reactor pressure to scram the control rods and the low probability of a DBA or transient occurring while the affected accumulators are inoperable.

C.1 and C.2

With one or more control rod scram accumulators inoperable and the reactor steam dome pressure < 900 psig, the pressure supplied to the charging water header must be adequate to ensure that accumulators remain charged. With the reactor steam dome pressure < 900 psig, the function of the accumulators in providing the scram force becomes much more important since the scram function could become severely degraded during a depressurization event or at low reactor pressures. Therefore, immediately upon discovery of charging water header pressure < 940 psig, concurrent with Condition C, all control rods associated with inoperable accumulators must be verified to be fully inserted. Withdrawn control rods with inoperable scram accumulators may fail to scram under these low pressure conditions. The associated control rods must also be declared inoperable within 1 hour. The allowed Completion Time of 1 hour is reasonable for Required Action C.2, considering the low probability of a DBA or transient occurring during the time the accumulator is inoperable.

D.1

The reactor mode switch must be immediately placed in the shutdown position if either Required Action and associated Completion Time associated with loss of the CRD pump (Required Actions B.1 and C.1) cannot be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the Required Action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1050 psig to 1100 psig (Ref. 2). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCES

1. USAR, Section 4.3.2.5.1.
2. USAR, Section 4.6.1.1.2.
3. USAR, Section 5.2.2.2.3.
4. USAR, Section 15.4.1.
5. 10 CFR 50.36(c)(2)(ii).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences effectively limit the potential amount of reactivity addition that could occur in the event of a control rod drop accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1, 2, and 3.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, and 3. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage, which could result in undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 4), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage, which would result in undue release of radioactivity (Refs. 5 and 6). Generic evaluations (Refs. 7 and 8) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 9) and the calculated offsite doses will be well within the required limits (Ref. 6).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS) described in Reference 10. The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are defined to minimize the maximum incremental control rod worths without being overly restrictive during normal plant operation. The generic BPWS analysis (Ref. 10) also evaluated the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

Rod pattern control satisfies the requirements of Criterion 3 of Reference 11.

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is ≤ 10% RTP, the CRDA is a Design Basis Accident (DBA) and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > 10% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, action may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to ≤ 10% RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence.

Required Action A.1 is modified by a Note, which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator (Reactor Operator or Senior Reactor Operator) or by a qualified member of the technical staff (e.g., a qualified shift technical advisor or reactor engineer). This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note that allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

a second licensed operator (Reactor Operator or Senior Reactor Operator) or by a qualified member of the technical staff (e.g., a qualified shift technical advisor or reactor engineer).

With nine or more OPERABLE control rods not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the reactor mode switch in shutdown, the reactor is shut down, and therefore does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

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

REFERENCES

1. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2, (revision specified in the COLR).

2. Letter from T.A. Pickens (BWROG) to G.C. Laines (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1988.

3. USAR, Section 15.4.9 and Appendix A.15.4.9.

4. NUREG-0979, "NRC Safety Evaluation Report for GESSAR II BWR/6 Nuclear Island Design, Docket No. 50-447," Section 4.2.1.3.2, April 1983.

(continued)

BASES

REFERENCES (continued)

5. NUREG-0800, "Standard Review Plan," Section 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2, July 1981.

6. 10 CFR 100.11, "Determination of Exclusion Area Low Population Zone and Population Center Distance."

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

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

9. ASME, Boiler and Pressure Vessel Code.

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

11. 10 CFR 50.36(c)(2)(ii).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

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

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, which are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the high pressure core spray system sparger.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System can also be automatically initiated as required by Reference 1; however, this is not necessary for SLC System OPERABILITY. The SLC System is used in the event that not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject, using both SLC pumps, a quantity of boron that produces a concentration of 660 ppm of natural boron in the reactor core, including recirculation loops, at 68°F and reactor water level at level 8. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 20% of the amount cited above is added (Ref. 2). An additional amount is provided to accommodate dilution in the RPV by the residual heat removal shutdown cooling piping. The volume versus concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved. This quantity of borated solution is the amount that is above the pump suction

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

The SLC System satisfies Criterion 4 of Reference 3.

LCO

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

APPLICABILITY

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

ACTIONS

A.1

If one SLC System subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to shutdown the unit. However, the overall capability is reduced since the remaining OPERABLE subsystem cannot meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of shutting down the unit and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive System to shut down the plant.

(continued)

BASES

ACTIONS (continued)

B.1

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

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances, verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring the SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure the proper borated solution and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The 24 hour Frequency of these SRs is based on operating experience that has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.4 and SR 3.1.7.6 (continued)

must be followed. The 31 day Frequency is based on operating experience that has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.6 verifies each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power operated, and automatic valves in the SLC System flow path ensures that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position, provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This Surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct positions. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensure correct valve positions.

SR 3.1.7.5

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure the proper concentration of boron (measured in weight % sodium pentaborate decahydrate) exists in the storage tank. SR 3.1.7.5 must be performed anytime boron or water is added to the storage tank solution to establish that the boron solution concentration is within the specified limits. This Surveillance must be performed anytime the temperature is restored to within the limit (i.e., $\geq 70°F$), to ensure no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.7

Demonstrating each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1235 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve, and is indicative of overall performance. Such inservice tests confirm component OPERABILITY and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 48 months, at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. While these Surveillances can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillances 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.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction valve to the injection pumps is unblocked ensures that there is a functioning flow

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.8 and SR 3.1.7.9 (continued)

path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping up to the suction valve is unblocked is to pump from the storage tank to the test tank. Upon completion of this verification, the pump suction piping between the pump suction valve and pump suction must be drained and flushed with demineralized water, since this piping is not heat traced. The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the daily temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.9 must be performed once within 24 hours after the piping temperature is restored within the limits of SR 3.1.7.3.

REFERENCES

1. 10 CFR 50.62.
2. USAR, Section 9.3.5.3.
3. 10 CFR 50.36(c)(2)(ii).

B 3.1 REACTIVITY CONTROL SYSTEMS

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

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all the control rods are capable of scramming. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2); and

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

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 (Ref. 2) and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves also allow continuous drainage of the SDV during normal plant

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

operation to ensure the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level exceeds a specified setpoint. The setpoint is chosen such that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of Reference 4.

LCO

The OPERABILITY of all SDV vent and drain valves ensures that, during a scram, the SDV vent and drain valves will close to contain reactor water discharged to the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to be open to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

In MODES 1 and 2, scram may be required, and therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram.

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each SDV vent and drain line. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions.

(continued)

BASES

ACTIONS (continued)

The ACTIONS Table is modified by a second Note stating that an isolated line may be unisolated under administrative control to allow draining and venting of the SDV. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable, since the administrative controls ensure the valve can be closed quickly, by a dedicated operator at the valve controls, if a scram occurs with the valve open.

A.1

When one SDV vent or drain valve is inoperable in one or more lines, the line must be isolated to contain the reactor coolant during a scram. The 7 day Completion Time is reasonable, given the level of redundancy in the lines and the low probability of a scram occurring during the time the valve(s) are inoperable and the line(s) not isolated. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion may not be preserved, and a higher risk exists to allow reactor water out of the primary system during a scram.

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

C.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended function during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation, which ensure correct valve positions. Improper valve position (closed) would not affect the isolation function.

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The 92 day Frequency is based on operating experience and takes into account the level of redundancy in the system design.

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of 30 seconds after a receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis. Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3, "Control Rod OPERABILITY," overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.8.3 (continued) the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
REFERENCES	1. USAR, Section 4.6.1.1.2. 2. 10 CFR 100. 3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981. 4. 10 CFR 50.36(c)(2)(ii).

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 and that the limits specified in 10 CFR 50.46 are not exceeded during the postulated design basis loss of coolant accident (LOCA). As a result, core geometry will be maintained following a design basis LOCA.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in Reference 1. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) and normal operations that determine APLHGR limits are presented in USAR, Chapters 4, 6, 15 and Appendix A, and in References 1, 2, and 3.

LOCA analyses are performed to ensure that the specified APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 1. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its 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 single recirculation loop operation, a conservative multiplier is applied to the exposure dependent APLHGR limits for two loop operation (Ref. 2). This limit 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The APLHGR satisfies Criterion 2 of Reference 4.
LCO	The APLHGR limits specified in the COLR are the result of fuel design and DBA analyses. For two recirculation loops operating, the limit is dependent on bundle 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. 2).
APPLICABILITY	The APLHGR limits are primarily derived from fuel design evaluations and LOCA 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 (IRM) 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 operates with substantial margin to the APLHGR limits; thus, this LCO is not required.
ACTIONS	<u>A.1</u> If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA analyses may not be met. Therefore, prompt action is taken to restore the APLHGR(s) to within the required limits such that the plant will be operating within analyzed conditions and within the design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a DBA occurring simultaneously with the APLHGR out of specification.

(continued)

BASES

ACTIONS (continued)	B.1 If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.2.1.1 APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution under normal conditions. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.
REFERENCES	1. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR). 2. USAR, Chapter 15B. 3. USAR, Chapter 15G. 4. 10 CFR 50.36(c)(2)(ii).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in the USAR, Chapters 4, 6, 15 and Appendix A, and References 2 and 3. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in USAR, Chapter 15B.

Flow dependent MCPR limits are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code and the multichannel thermal hydraulic code (Ref. 3). The worst flow increase transient results from recirculation flow controller failure. The K_f curve is derived assuming both recirculation loop controllers fail. This condition produces the maximum possible power increase and hence maximum ΔCPR for transients initiated from less than rated power and flow.

The MCPR satisfies Criterion 2 of Reference 4.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limit is determined by multiplying the normal MCPR operating limit (100% core flow) by a correction factor based on actual core flow.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

Statistical analyses documented in Reference 5 indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2

(continued)

BASES

APPLICABILITY (continued)

occurs. When in MODE 2, the intermediate range monitor (IRM) provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

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

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER reaches ≥ 25% RTP is acceptable given the large inherent margin to operating limits at low power levels.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.
2. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).
3. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2, (revision specified in the COLR).
4. 10 CFR 50.36(c)(2)(ii).
5. "BWR/6 Generic Rod Withdrawal Error Analysis," General Electric Standard Safety Analysis Report, GESSAR-II, Appendix 15B.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LHGR satisfies Criterion 2 of Reference 4.

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared with the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution under normal conditions. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

1. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).

2. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2, (revision specified in the COLR).

3. NUREG-0800, Section II A.2(g), Revision 2, July 1981.

4. 10 CFR 50.36(c)(2)(ii).

B 3.2 POWER DISTRIBUTION LIMITS

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

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design"; GDC 13, "Instrumentation and Control"; GDC 20, "Protection System Functions"; and GDC 29, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b) to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP) where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{\text{MFLPD}}{\text{FRTP}} > 1,$$

indicating that MFPLD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by gain adjustment on the APRMs or adjustment of the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR, MCPR, and LHGR (Ref. 3).

The normally selected APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line, such that an

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BASES

BACKGROUND (continued)

approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram or rod blocks is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM Allowable Value is supported by the analyses presented in Reference 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

The APRM neutron flux signal is also adjusted to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant less than, but approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained, while a conservative margin is maintained. The delayed response of the filtered APRM signal allows the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.

APPLICABLE SAFETY ANALYSES

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

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BASES

APPLICABLE SAFETY ANALYSES (continued)

USAR safety analyses (Ref. 2) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, MCPR, and LHGR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of the APLHGR, the MCPR, or the LHGR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pretransient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value is required to be reduced by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value dependent on the increased peaking that may be encountered.

The APRM gain and setpoint satisfy Criteria 2 and 3 of Reference 4.

LCO

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

a. Limiting excess power peaking;

b. Reducing the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value by multiplying the APRM Flow Biased Simulated Thermal Power—Upscale

(continued)

BASES

LCO (continued)

Function Allowable Value by the ratio of FRTP to the core limiting value of MFLPD; or

c. Increasing the APRM gains to cause the APRM to read greater than 100(%) times MFLPD. This Condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value is reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value. Adjusting the APRM gain or modifying the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value is equivalent to maintaining MFLPD less than or equal to FRTP, as stated in the LCO.

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

APPLICABILITY

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

(continued)

BASES (continued)

ACTIONS

A.1

If the APRM gain or Flow Biased Simulated Thermal Power—Upscale Function Allowable Value is not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or Flow Biased Simulated Thermal Power—Upscale Function Allowable Value to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

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

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared to FRTP or APRM gain or Flow Biased Simulated Thermal Power—Upscale Function Allowable Value to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are required only to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate APRM gain or Flow Biased Simulated Thermal Power—Upscale Function Allowable Value, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or Flow Biased Simulated Thermal Power—Upscale Function circuitry. SR 3.2.4.1 and SR 3.2.4.2 have been modified by Notes, which

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ASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

clarify that the respective SR does not have to be met if the alternate requirement demonstrated by the other SR is satisfied. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1) and LHGR (LCO 3.2.3). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 29.
2. USAR, Chapter 15 and Appendix A.
3. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).
4. 10 CFR 50.36(c)(2)(ii).