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April 26, 2002
JAFP-02-0098

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-59

Revision J to Proposed Technical Specification Change (License Amendment) Conversion to Improved Standard Technical Specifications

References: see last page of letter

Dear Sir,

This letter and the associated attachments provides Revision J to the previously submitted application for amendment to the James A. FitzPatrick Technical Specifications (Reference 1), as supplemented by References 2, 3, 4, and 5 for converting the current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) consistent with the Improved Standard Technical Specifications (NUREG-1433, Revision 1).

Revision J (Attachment 1) to the Reference 1, 2, 3, 4 , and 5 submittals include: certain Technical Specification Task Force Traveler related changes; a change to close out a remaining NRC question; numerous typographical, editorial, and consistency corrections; changes due to the engineering analysis performed as discussed in Reference 6; and a few new additional changes. Each Chapter/Section includes a summary of the changes to the associated Chapter/Section (with the exception of the Split Report, whose summary for the change is included in the Summary of Changes to Section 3.7).

The Insert and Discard Instructions are included in Attachment 2 to allow merging Revision J with the existing submittal. The clean typed ITS and Bases in Volumes 2, 3, and 4, and the CTS markup pages in CTS order in Volume 5 are not being updated since these Volumes are duplicates of each individual Specification located in Volumes 6 through 19.

We request that you approve the James A. FitzPatrick ITS no later than July 31, 2002.

A001

There are no new commitments contained in this letter. Should you have any questions, please contact Mr. Andrew Halliday at (315) 349-6055.

Very Truly Yours,

T. A. Sullivan
Vice President, Operations - JAF

Attachments: 1) Revision J to the JAF ITS Submittal
2) Insert and Discard Instructions

cc:

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References:

1. NYPA letter, J. Knubel to USNRC Document Control Desk, Proposed Technical Specification Change (License Amendment) Conversion to Improved Standard Technical Specifications (JPN-99-008), dated March 31, 1999 (TAC No. MA5049)

2. NYPA letter, J. Knubel to USNRC Document Control Desk, Revision B to Proposed Technical Specification Change (License Amendment) Conversion to Improved Standard Technical Specifications (JPN-99-018), dated June 1, 1999

3. NYPA letter, Michael J. Colomb to USNRC Document Control Desk, Revision C to Proposed Technical Specification Change (License Amendment) Conversion to Improved Standard Technical Specifications (JAFP-99-0278), dated October 14, 1999

4. Entergy Nuclear Northeast letter, T. A. Sullivan to USNRC Document Control Desk, Revisions D, E, F, G, and H to Proposed Technical Specification Change (License Amendment) Conversion to Improved Standard Technical Specifications (JAFP-01-0133), dated May 31, 2001

5. Entergy Nuclear Northeast letter, T. A. Sullivan to USNRC Document Control Desk, Revision I to Proposed Technical Specification Change (License Amendment) Conversion to Improved Standard Technical Specifications (JAFP-01-0234), dated October 18, 2001

6. Entergy Nuclear Northeast letter, T. A. Sullivan to USNRC Document Control Desk, James A. FitzPatrick (JAF) Improved Technical Specifications (ITS) Submittal (JAFP-02-0029), dated February 6, 2002

BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
Entergy Nuclear Operations, Inc.	)	Docket No. 50-333
James A. FitzPatrick Nuclear Power Plant	)	

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Entergy Nuclear Operations, Inc. requests an amendment to the Technical Specifications (TS) contained in Appendix A to Facility Operating License DPR-59 for the James A. FitzPatrick Nuclear Power Plant. This application is filed in accordance with Section 10 CFR 50.90 of the Nuclear Regulatory Commission's regulations.

This application for amendment to the FitzPatrick Technical Specifications proposes to convert the FitzPatrick current Technical Specifications (CTS) to be consistent with the Improved Standard Technical Specifications (ISTS) in NUREG-1433, Revision 1, dated April 1995. The proposed license amendment request was prepared considering the guidance of Nuclear Energy Institute (NEI) NEI 96-06, "Improved Technical Specifications Conversion Guidance," dated August 1996.

The Proposed license amendment request to convert the FitzPatrick CTS to the FitzPatrick Improved Technical Specifications (ITS) is enclosed with this application.

Entergy Nuclear Operations, Inc.

T. A. Sullivan
Vice President, Operations-JAF

STATE OF NEW YORK
COUNTY OF OSWEGO
Subscribed and sworn to before me
this 26 day of April 2002.

Nancy B. Czerow
Notary Public

NANCY B. CZEROW
Notary Public, State of New York
Qualified in Oswego County [illegible]
Commission Expires 1-26-03

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
Retyped ITS typographical errors	Minor typographical errors in the retyped ITS have been corrected to be consistent with the NUREG markup. (The SR 3.6.1.2.1 word "NOTE" has been changed to "NOTES"; the word "System" has been added after the word "LPCI" in ITS 3.6.1.3 Condition E; the word "System" has been changed to "Systems" in SR 3.6.1.3.11; the word "subsystem" has been changed to "subsystems" in ITS 3.6.1.8 Condition B; the period at the end of LCO 3.6.2.1.a has been changed to a semicolon; the ITS 3.6.4.2 Required Action A.2 word "NOTE" has been changed to "NOTES"; and a comma has been added after the word "operated" in SR 3.6.4.2.2.)	**Specification 3.6.1.2** Retyped ITS p 3.6-6 **Specification 3.6.1.3** Retyped ITS p 3.6-11 and 3.6-14 **Specification 3.6.1.8** Retyped ITS p 3.6-22 **Specification 3.6.2.1** Retyped ITS p 3.6-26 **Specification 3.6.4.2** Retyped ITS p 3.6-40 and 3.6-42
NUREG ITS markup errors	Minor NUREG markup errors have been corrected to be consistent with the retyped ITS. (A comma has been added to the SR 3.6.1.3.1 Note; the words "THERMAL POWER is > 1% RTP" have been changed to "THERMAL POWER > 1% RTP" in LCO 3.6.2.1.a and b; and the words "THERMAL POWER is ≤ 1% RTP" have been changed to "THERMAL POWER ≤ 1% RTP" in LCO 3.6.2.1.c.)	**Specification 3.6.1.3** NUREG ITS markup p 3.6-14 **Specification 3.6.2.1** NUREG ITS markup p 3.6-31

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
Retyped ITS Bases typographical errors	Minor typographical errors in the retyped ITS Bases have been corrected to be consistent with the NUREG Bases markup. (The word "Rate" has been added to the ITS 3.6.1.1 LCO section; the words "to meet" have been changed to "of" in the SR 3.6.1.1.1 section; the words "entering and exiting the drywell" have been changed to "entry and exit", a comma has been added, and a comma has been deleted in the ITS 3.6.1.2 Background section; the words "design basis LOCA maximum" have been added to the ITS 3.6.1.2 ASA section; the word "rates" and "into" have been added to the ITS 3.6.1.2 LCO section; the words "after each entry and exit" have been moved to the beginning of the sentence and a comma added, the word "actions" has been changed to "ACTIONS", a comma has been deleted (two places), and the word "lock(s)" has been changed to "lock" in the ITS 3.6.1.2 Actions section; the word "respect" has been changed to "regard" and the word "rate" has been added to the SR 3.6.1.2.1 section; a comma has been added to the ITS 3.6.1.3 Actions A.1 and A.2 section; the words "except for inoperabilities due to MSIV, LPCI or CS System air operated testable check valves leakage not within limits" have been added, the last sentence of the second paragraph has been deleted, and the word "valves" has been changed to "devices" in the ITS 3.6.1.3 Actions C.1 and C.2 section; the words "System" and "air operated" have been added and the word "limit" has been changed to "limits" in the ITS 3.6.1.3 Actions E.1 section; the words "12 inch" have been added and the word "that" has been changed to "the" in the SR 3.6.1.3.1 section; the word "PCIVs" has been changed to "isolation devices" (four places), a comma has been deleted, the word "and" has been added, and the words "or equivalent isolation methods" have been deleted from the third paragraph in the SR 3.6.1.3.2 section; the word "PCIVs" has been changed to "isolation devices" (three places), a comma has been deleted, the word "and" has been added, and the words "and equivalent isolation methods" have been deleted from the second paragraph in the SR 3.6.1.3.3 section; "Reference 8" has been changed to "Reference 10" in the SR 3.6.1.3.10 section; the word "rate" has been added to the SR 3.6.1.3.11 section; periods have been added to the ITS 3.6.1.3 References 4, 5, and 6; the word "pressuure" has been changed to "pressure" in the ITS 3.6.1.5 ASA section; the words "heating and" have been deleted from the ITS 3.6.1.6 Background section; the word "also" has been added to the ITS 3.6.1.6 Applicability section; the word "the" has been added (two places) to the ITS 3.6.1.7 Background section; the words "during testing or" have been added and the words "also are" have been changed to "are also" in the ITS 3.6.1.7 LCO section; the word "test" has been deleted in the ITS 3.6.1.7 Actions B.1 section; the value ".75" has been changed to "0.75" in the ITS 3.6.1.9 ASA section; the word "(S/RV)" has been deleted in the ITS 3.6.2.1 Background section; the sentence "An adequate average is obtained if at least 15 of the bays are monitored." has been deleted and a comma added after the LCO title in the SR 3.6.2.1.1 section; a dash has been added between "GE" and "NE" in ITS 3.6.2.1 Reference 3; (CONTINUED ON NEXT PAGE)	**Specification 3.6.1.1** Retyped ITS Bases B p 3.6-3 and B 3.6-4 **Specification 3.6.1.2** Retyped ITS Bases B p 3.6-6, B 3.6-7, B 3.6-8, B 3.6-9, B 3.6-11, and B 3.6-12 **Specification 3.6.1.3** Retyped ITS Bases p B 3.6-18, B 3.6-20, B 3.6-21, B 3.6-22, B 3.6-23, B 3.6-24, B 3.6-25, B 3.6-27, and B 3.6-28 **Specification 3.6.1.5** Retyped ITS Bases p B 3.6-32 **Specification 3.6.1.6** Retyped ITS Bases p B 3.6-35 and B 3.6-37 **Specification 3.6.1.7** Retyped ITS Bases p B 3.6-42, B 3.6-43, B 3.6-44, and B 3.6-45 **Specification 3.6.1.9** Retyped ITS Bases p B 3.6-53 **Specification 3.6.2.1** Retyped ITS Bases p B 3.6-57 and B 3.6-62

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
Retyped ITS Bases typographical errors (continued)	the words "system downcomer" have been added to the ITS 3.6.2.2 ASA section; the word "operability" has been changed to "OPERABILITY" in the ITS 3.6.2.2 LCO section; the short dash in the LCO 3.5.2 title has been changed to a long dash in ITS 3.6.2.2 Applicability section; the word "loops" has been changed to "subsystems" in the ITS 3.6.2.3 Background section; the word "indicates" has been changed to "indicate" in the ITS 3.6.2.3 ASA section; the words "(Ref. 2)" have been changed to "(Ref. 3)" in the ITS 3.6.2.3 LCO section; one paragraph has been split into two paragraphs in the ITS 3.6.3.2 Applicability section; numerous changes to the SR 3.6.4.1.4 section have been made to match the NUREG Bases markup; one paragraph has been split into two paragraphs in the ITS 3.6.4.2 Actions B.1 section; and the last paragraph of the SR 3.6.4.3.4 section has been deleted.)	**Specification 3.6.2.2** Retyped ITS Bases p B 3.6-64 **Specification 3.6.2.3** Retyped ITS Bases p B 3.6-67 and B 3.6-68 **Specification 3.6.3.2** Retyped ITS Bases p B 3.6-81 **Specification 3.6.4.1** Retyped ITS Bases p B 3.6-88 and B 3.6-89 **Specification 3.6.4.2** Retyped ITS Bases p B 3.6-94 **Specification 3.6.4.3** Retyped ITS Bases p B 3.6-103

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
NUREG Bases markup errors	Minor NUREG Bases markup errors have been corrected to be consistent with the retyped ITS Bases. (The word "Rate" has been added to SR 3.6.1.1.1; the word "unit" has been changed to "plant" in the ITS 3.6.1.2 Background section; the word "the" has been deleted from the ITS 3.6.1.2 ASA section; a period has been added to ITS 3.6.1.2 Reference 2; the words "valves 27MOV-120 and 27MOV-121" have been changed to "these valves" in the ITS 3.6.1.3 LCO section; the word "cintainment" has been changed to "containment" in ITS 3.6.1.3 INSERT ASA-1; the word "inoperabilies" has been changed to "inoperabilities" in the ITS 3.6.1.3 Actions A.1 and A.2 section; the word "consodered" has been changed to "considered" in the ITS 3.6.1.3 INSERT ACTIONS ASA-1; the words "isolated, the leakage rate for the isolated penetration is" has been added to ITS 3.6.1.3 INSERT ACTION E; the SR number has been changed from "3.6.1.3.3" to "3.6.1.3.2" in the SR 3.6.1.3.1 section; the word "the" has been changed to "this" in the SR 3.6.1.3.8 section; the period has been deleted in INSERT SR 3.6.1.3.11; a comma has been added to ITS 3.6.1.3 Reference 9; a period has been added to ITS 3.6.1.3 Reference 12; a period has been added to ITS 3.6.1.5 Reference 4; a comma has been added to the ITS 3.6.1.6 Background section; the word "withthe" has been changed to "with the" in the ITS 3.6.1.6 Insert SR 3.6.1.6.3; the word "suppress" has been changed to "suppression" in the ITS 3.6.1.7 ASA section; periods have been added to the ITS 3.6.1.7 References 3, 4, and 5; a period has been added to ITS 3.6.1.8 Reference 3; the words "RATED THERMAL POWER (RTP)" have been changed to "RTP"; periods have been added to ITS 3.6.2.1 References 2, 3, and 5; periods have been added to ITS 3.6.2.2 References 2 and 3; a comma has been added to the ITS 3.6.2.3 Actions C.1 and C.2 section; periods have been added to ITS 3.6.2.4 References 1 and 2; (CONTINUED ON NEXT PAGE)	**<u>Specification 3.6.1.1</u>** NUREG Bases markup p B 3.6-4 **<u>Specification 3.6.1.2</u>** NUREG Bases markup p B 3.6-6, B 3.6-7, and B 3.6-14 **<u>Specification 3.6.1.3</u>** NUREG Bases markup p B 3.6-15, Insert Page B 3.6-16, B 3.6-18, Insert Page B 3.6-20, Insert Page B 3.6-22, B 3.6-25, B 3.6-29, Insert Page B 3.6-31, and B 3.6-32 **<u>Specification 3.6.1.5</u>** NUREG Bases markup p B 3.6-37 **<u>Specification 3.6.1.6</u>** NUREG Bases markup p B 3.6-43 and Insert Page B 3.6-47 **<u>Specification 3.6.1.7</u>** NUREG Bases markup p B 3.6-49 and B 3.6-53 **<u>Specification 3.6.1.8</u>** NUREG Bases markup p B 3.6-57 **<u>Specification 3.6.2.1</u>** NUREG Bases markup p B 3.6-59 and B 3.6-63 **<u>Specification 3.6.2.2</u>** NUREG Bases markup p B 3.6-66 **<u>Specification 3.6.2.3</u>** NUREG Bases markup p B 3.6-69 **<u>Specification 3.6.2.4</u>** NUREG Bases markup p B 3.6-77

Source of Change	Summary of Change	Affected Pages
NUREG Bases markup errors (continued)	a comma has been added to ITS 3.6.3.2 Reference 3; a period has been added to ITS 3.6.4.2 Reference 4; the word "operability" has been changed to "OPERABILITY" in the SR 3.6.4.3.3 section; a comma has been added to ITS 3.6.4.3 Reference 1; and a period has been added to ITS 3.6.4.3 Reference 4.)	**Specification 3.6.3.2** NUREG Bases markup p B 3.6-96 **Specification 3.6.4.2** NUREG Bases markup p B 3.6-108 **Specification 3.6.4.3** NUREG Bases markup p B 3.6-114
Typographical errors	Minor typographical errors have been corrected in the Discussion of Changes, NUREG ITS markup, NUREG Bases markup, retyped ITS, and retyped ITS Bases. (The word "Section" has been added between the words "UFSAR" and "7.3.3.1" in the ITS 3.6.1.3 ASA Bases section; the comma has been deleted after the word "MSIV" and the words "leakage or" have been added (3 places) in ITS 3.6.1.3 Actions A.1 and A.2, Action B.1, and Actions C.1 and C.2 Bases; the "<" sign has been changed to "<_" in ITS 3.6.1.6 DOC M3; the term "reactor-to-suppression chamber" vacuum breakers in SR 3.6.1.6.1 Bases has been changed to "reactor building-to-suppression chamber" vacuum breakers; the title "High Pressure Core Injection System" has been changed to "High Pressure Coolant Injection System" in the ITS 3.6.2.2 Bases LCO section; the word "Reference" has been changed to "References" (2 places) in the ITS 3.6.2.3 ASA Bases section; a period has been added to ITS 3.6.4.3 Required Action D.1.)	**Specification 3.6.1.3** NUREG Bases markup p Insert Page B 3.6-16, B 3.6-18, and B 3.6-20 Retyped ITS Bases p B 3.6-16, B 3.6-18, and B 3.6-20 **Specification 3.6.1.6** DOC M3 (DOCs p 1 of 5) NUREG Bases markup p B 3.6-47 Retyped ITS Bases p B 3.6-39 **Specification 3.6.2.2** NUREG Bases markup p Insert Page B 3.6-65 Retyped ITS Bases p B 3.6-64 **Specification 3.6.2.3** NUREG Bases markup p B 3.6-67 and Insert Page B 3.6-67 Retyped ITS Bases p B 3.6-67 **Specification 3.6.4.3** NUREG ITS markup p 3.6-55 Retyped ITS p 3.6-44

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
Consistency issues	Minor consistency issue corrections have been made. (The SR 3.6.1.1.2 Bases have been modified to include the information relocated by DOC LA3 (i.e., the drywell to suppression chamber leakage limit of 71 scfm); the words "reactor building" have been added to the SR 3.6.1.3.1 Bases to be consistent with the actual SR; the unit "inch" has been added after the value "20" in SR 3.6.1.3.1 and associated Note, since each value should have the unit immediately after it; a comma has been added after the word "operated" in SR 3.6.1.3.5 and associated Bases, consistent with TSTF-46, Rev. 1 and SR 3.6.4.2.2; the word "Five" has been changed to "Each" in LCO 3.6.1.7, since all suppression chamber-to-drywell vacuum breakers are required Operable (this is also consistent with LCO 3.6.1.6, which uses ""Each" in the LCO statement since all reactor building-to-suppression chamber vacuums breakers are required Operable); the word "required" has been deleted from SR 3.6.1.7.2 and SR 3.6.1.7.3 for consistency with the usage throughout the ITS, since all suppression chamber-to-drywell vacuum breakers are required Operable; quotation marks have been placed around the LCO 3.6.1.1 title in the ITS 3.6.1.7 Action B.1 Bases section and the title for LCO 3.6.1.1 has been deleted from SR 3.6.1.7.1 Bases since it is identified earlier in the Bases; SR 3.6.1.9 has been modified to be consistent with the similar SR in ITS 3.6.2.3 (SR 3.6.2.3.2) and to be consistent with the associated NUREG Bases (which says that the flow is through the associated heat exchanger while operating in the suppression pool cooling mode); (CONTINUED ON NEXT PAGE)	**Specification 3.6.1.1** NUREG Bases markup p B 3.6-4 Retyped ITS Bases p B 3.6-4 **Specification 3.6.1.3** NUREG ITS markup p 3.6-14 and 3.6-15 NUREG Bases markup p B 3.6-25 and B 3.6-27 Retyped ITS p 3.6-12 and 3.6-13 Retyped ITS Bases p B 3.6-23 and B 3.6-26 **Specification 3.6.1.7** NUREG ITS markup p 3.6-26 and 3.6-28 JFD DB2 (JFDs p 2 of 3) NUREG Bases markup p B 3.6-51 and B 3.6-52 Bases JFD DB7 (Bases JFDs p 3 of 5) Retyped ITS p 3.6-20 and 3.6-21 Retyped ITS Bases p B 3.6-45, B 3.6-46, and B 3.6-47 **Specification 3.6.1.9** NUREG ITS markup p Insert Page 3.6-30b JFD PA3 (JFDs p 1 of 2) Retyped ITS p 3.6-25

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
Consistency issues (continued)	the term "Specification 3.3.3.1" has been changed to "LCO 3.3.3.1" and quotation marks have been placed around the LCO title in the SR 3.6.2.1.1 Bases; the Notes to SR 3.6.2.2.1 and SR 3.6.2.4.1 have been moved to the LCOs, since the Notes actually modify the LCO statement (this is consistent with similar Notes in LCO 3.3.3.1, LCO 3.4.7, and LCO 3.4.8 and with the CTS), and the words "up to" have been added to the LCO 3.6.2.4 Note for consistency with the LCO 3.6.2.2 Note; the ITS 3.6.2.3 Bases Reference 4 "10 CFR 50.36 (c)(2)(ii)" has been changed to "10 CFR 50.36(c)(2)(ii)"; the term "ft" has been added after the value "0.36" in the ITS 3.6.2.4 ASA and LCO Bases sections; CONTINUED ON NEXT PAGE	**Specification 3.6.2.1** NUREG Bases markup p B 3.6-62 Retyped ITS Bases p B 3.6-62 **Specification 3.6.2.2** CTS markup p 1 of 3 DOC LA1 (DOCs p 2 of 3) NUREG ITS markup p 3.6-34 JFD CLB1 (JFDs p 1 of 1) NUREG Bases markup p B 3.6-65, Insert page B 3.6-65, and B 3.6-66 (Insert page B 3.6-66 deleted) Bases JFD CLB1 (Bases JFDs p 1 of 2) Retyped ITS p 3.6-29 Retyped Bases p B 3.6-64 and 65 **Specification 3.6.2.3** NUREG Bases markup p Insert Page B 3.6-70 Retyped ITS Bases p B 3.6-71 **Specification 3.6.2.4** CTS markup p 2 of 3 DOC LA1 (DOCs p 1 of 2) NUREG ITS markup p 3.6-39 and Insert Page 3.6-39 JFD CLB2 (JFDs p 1 of 1) NUREG Bases markup p B 3.6-75, B 3.6-76, Insert page B 3.6-76, and B 3.6-77 (Insert page B 3.6-77 deleted) Bases JFD CLB2 (Bases JFDs p 1 of 2) Retyped ITS p 3.6-32 and 3.6-33 Retyped Bases p B 3.6-72, B 3.6-73, and B 3.6-74

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
Consistency issues (continued)	and the title of LCO 3.6.4.3 has been deleted from Insert SR 3.6.4.1.4 since it is already listed in the background section.)	**Specification 3.6.4.1** NUREG Bases markup p Insert page B 3.6-101 Retyped Bases p B 3.6-89
Consistency issue	The Bases Background section of ITS 3.6.1.8 has been modified to more clearly state how the MSLC System operates, consistent with the UFSAR description.	**Specification 3.6.1.8** NUREG Bases markup p Insert Page B 3.6-54 Retyped ITS Bases p B 3.6-48 and B 3.6-49
Editorial change	Amplifying information regarding the overall air lock leakage rate limit has been added for completeness.	**Specification 3.6.1.2** NUREG Bases markup p B 3.6-12 and B 3.6-14 Bases JFD X2 (Bases JFDs p 3 of 3) Retyped ITS Bases p B 3.6-12 and B 3.6-13
Editorial change	The proper References and descriptions of the safety analyses assumptions have been provided for ITS Bases 3.6.1.3.	**Specification 3.6.1.3** NUREG Bases markup p B 3.6-15, B 3.6-16, Insert Page B 3.6-16, B 3.6-17, B 3.6-21, B 3.6-29, B 3.6-31, and B 3.6-32 Bases JFD DB4 (Bases JFDs p 3 of 6) Retyped ITS Bases p B 3.6-16, B 3.6-17, B 3.6-20, B 3.6-27, and B 3.6-28

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
Editorial change	The proper References have been provided for ITS 3.6.1.4, ITS 3.6.1.5, and ITS 3.6.1.7 Bases.	**Specification 3.6.1.4** NUREG Bases markup p B 3.6-33 and Insert Page B 3.6-34 Retyped ITS Bases p B 3.6-29 and B 3.6-31 **Specification 3.6.1.5** NUREG Bases markup p B 3.6-35, B 3.6-37, and Insert Page B 3.6-37 Retyped ITS Bases p B 3.6-32 and B 3.6-34 **Specification 3.6.1.7** NUREG Bases markup p B 3.6-49, Insert Page B 3.6-49, B 3.6-50, and B 3.6-53 Retyped ITS Bases p B 3.6-43, B 3.6-44, and B 3.6-47
Editorial change	In lieu of specifying specific zones (which are not currently used at FitzPatrick), the SR 3.6.1.5 Bases describes that temperature is monitored in "various areas" and at various elevations.	**Specification 3.6.1.5** NUREG Bases markup p B 3.6-36 Retyped ITS Bases p B 3.6-33
Editorial change	The proper document number has been provided for ITS 3.6.2.1 Reference 3 (i.e., "0737" has been changed to "00737"), ITS 3.6.2.2 Reference 2 (i.e., "0737" has been changed to "00737" and the term "RHR" has been added after the word "Higher"), and ITS 3.6.2.3 Reference 2 (i.e., "0737" has been changed to "00737" and the term "RHR" has been added after the word "Higher").	**Specification 3.6.2.1** NUREG Bases markup p B 3.6-63 Retyped ITS Bases p B 3.6-62 **Specification 3.6.2.2** NUREG Bases markup p B 3.6-66 Retyped ITS Bases p B 3.6-66 **Specification 3.6.2.3** NUREG Bases markup p Insert Page B 3.6-70 Retyped ITS Bases p B 3.6-70
Editorial change	The equivalent downcomer waterleg value of 0.36 ft in the ITS 3.6.2.4 Bases Applicable Safety Analyses and LCO section has been rounded up to 0.37 ft.	**Specification 3.6.2.4** NUREG Bases markup p B 3.6-75 and B 3.6-76 Retyped ITS Bases p B 3.6-72 and B 3.6-73

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
Editorial change	SR 3.6.4.2.2 has been modified to use the second Frequency provided in the NUREG; "92 days" in lieu of "In accordance with the Inservice Testing Program." This is necessary since the secondary containment isolation dampers are not part of the IST Program at JAFNPP. In addition, the stroke times will also be listed in the TRM.	**Specification 3.6.4.2** DOC M8 (DOCs p 5 of 10) NUREG ITS markup p 3.6-53 JFD X1 (JFDs p 1 of 1) NUREG Bases markup p B 3.6-103 and B 3.6-107 Bases JFDs X1 and X3 (Bases JFDs p 2 of 2) Retyped ITS p 3.6-42 Retyped Bases p B 3.6-91 and B 3.6-96
Editorial change	The parenthetical phrase in the ITS 3.6.4.3 Bases Background section (200% of reactor building free volume per day) has been deleted since it is not necessary to include in the Bases. Also, a parenthetical phrase has been modified for clarity in the Bases for SR 3.6.4.3.3 (the phrase "an interlock with suction valve" has been changed to "interlocked with the suction valve").	**Specification 3.6.4.3** NUREG Bases markup p B 3.6-109 and B 3.6-114 Retyped ITS Bases p B 3.6-98 and B 3.6-102
Technical change	The leakage limit for the pneumatic test of the air operated testable check valves in the CS and LPCI Systems has been reduced from 11 scfm to 10 scfm, consistent with the NRC Safety Evaluation Report for Technical Specification Amendment 40, dated November 9, 1978.	**Specification 3.6.1.3** CTS markup page 7 of 10 DOCs M6 and LA1 (DOCs p 6 of 14 and 7 of 14) JFD CLB11 (JFDs p 2 of 5) NUREG Bases markup p B 3.6-31 Retyped ITS Bases p B 3.6-28

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
Technical change	The suppression chamber-to-drywell and reactor building-to-suppression chamber vacuum breakers are not designed or assumed to be fully open at 0.5 psid. They are only required to start to open at 0.5 psid. Therefore, ITS 3.6.1.6 and ITS 3.6.1.7 have been changed to adopt the NUREG words in the functional test SRs that test the opening setpoint (Note: SR 3.6.1.7.3 already reflects the NUREG words), and the associated Bases have been modified to describe that "opening setpoint" means the point at which the vacuum breakers start to open.	**Specification 3.6.1.6** DOC M3 (DOCs p 1 of 5) NUREG ITS markup p 3.6-25 JFD DB3 (deleted) (JFDs p 1 of 2) NUREG Bases markup p Insert page B 3.6-44 and B 3.6-47 Retyped ITS p 3.6-19 Retyped ITS Bases p B 3.6-36 and B 3.6-40 **Specification 3.6.1.7** NUREG Bases markup p B 3.6-49 and B 3.6-53 Retyped ITS Bases p B 3.6-43 and B 3.6-47
Technical change	The proper containment temperature envelope has been provided in the ASA Bases for ITS 3.6.1.9, based on the most recent analysis (the temperature changed from 330 degrees F to 335 degrees F and the time changed from 200 seconds to 300 seconds), and a description of the safety analysis has been provided. In addition, the proper References have been provided.	**Specification 3.6.1.9** NUREG Bases markup p Insert Page B 3.6-57c, Insert Page B 3.6-57d, Insert Page B 3.6-57f, Insert Page B 3.6-57g, and Insert Page B 3.6-57h Retyped ITS Bases p B 3.6-53, B 3.6-55, and B 3.6-56
Technical change	The safety analyses description of ITS 3.6.2.1 Bases has been modified to be consistent with the current safety analyses. Also, the proper References have been provided.	**Specification 3.6.2.1** NUREG Bases markup p B 3.6-59, Insert Page B 3.6-59, B 3.6-60, and B 3.6-63 Bases JFD PA5 (deleted) (Bases JFDs p 1 of 2) Retyped ITS Bases p B 3.6-57, B 3.6-58, B 3.6-59, and B 3.6-62

SUMMARY OF CHANGES TO ITS SECTION 3.6 - REVISION J

Source of Change	Summary of Change	Affected Pages
TSTF-30 and TSTF-207	A recent review by JAFNPP personnel of the primary containment penetration list has shown that some penetrations have more that two PCIVs, and some penetrations that are not in a closed system that have a single valve that is not an EFCV. Therefore, changes have been made to adopt the relevant portions of TSTF-30 and TSTF-207.	**Specification 3.6.1.3** DOCs A3, L3, L4, L5, and L10 (DOCs p 2 of 14, 9 of 14, 10 of 14, and 12 of 14) NSHC L4 (NSHCs p 5 of 22) NUREG ITS markup p 3.6-8 and 3.6-10 JFD TA3 (JFDs p 3 of 5) NUREG Bases markup p B 3.6-14, B 3.6-19, B 3.6-20, Insert Page B 3.6-20, and B 3.6-21 Retyped ITS p 3.6-8, 3.6-9, and 3.6-10 Retyped ITS Bases p B 3.6-14, B 3.6-19, B 3.6-20, and B 3.6-21

Failure to meet the Low Pressure Coolant Injection (LPCI) or Core Spray (CS) System injection line air operated testable check valve leakage limit (SR 3.6.1.3.11) does not result in failure of this SR since the LPCI and CS testable check valve leakage is not included in the Primary Containment Leakage Rate Testing Program limits (Ref. 5 and 6)

CLB4

BASES (continued) Rate

1 J

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

PA2 — the — PA2 — the — X3 — 10 — PA1 — limit — PA2 — the — PA2 — PA2

Primary Containment Leakage Rate Testing Program

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

the Primary Containment Leakage Rate Testing Program

TA1

≤

SR 3.6.1.1.2

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR ~~measures drywell to suppression chamber differential pressure during a [10] minute period to ensure that the~~ leakage paths that would bypass the suppression pool are within allowable limits (i.e., ≤ 71 scfm).

is a leak test that confirms that the bypass area between the drywell and suppression chamber is less than the equivalent of a one inch diameter plate orifice (Ref. 4). This ensured that the

DB4 — DB4 — DB4 — (≥ 1 psi)

1 J

Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and verifying that the pressure in ~~either~~ the suppression chamber ~~or the drywell~~ does not ~~change~~ increase by more than [0.25] inch of water per minute over a 10 minute period. The leakage test is performed every ~~[18 months]~~ 24 months. The ~~[18~~ 24 month~~]~~ Frequency ~~was~~ ~~developed~~

increase — 24 — CLB2 — es — PA2

(continued)

Revision J

BASES

LCO (continued)

leakage to those leakage rates assumed in the safety analyses.

Individual leakage rates for the primary containment air locks are addressed in LCO 3.6.1.2 and specified in the Primary Containment Leakage Rate Testing Program. 5

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

ACTIONS

A.1

In the event primary containment is inoperable, primary containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining primary containment OPERABILITY during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring primary containment OPERABILITY) occurring during periods where primary containment is inoperable is minimal.

B.1 and B.2

If primary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. Failure to meet the air lock leakage limit (SR 3.6.1.2.1), or the main steam isolation valve leakage limit (SR 3.6.1.3.10) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program. Failure to meet the Low Pressure Coolant Injection (LPCI) or Core Spray (CS)System injection line air operated testable check valve leakage limit (SR 3.6.1.3.11) does not result in failure of this SR since the LPCI and CS testable check valve leakage is not included in the Primary Containment Leakage Rate Testing Program limits (Ref. 5 and 6).

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

SR 3.6.1.1.2

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR is a leak test that confirms that the bypass area between the drywell and suppression chamber is less than the equivalent of a one inch diameter plate orifice (Ref. 1). This ensures that the leakage paths that would bypass the suppression pool are within allowable limits (i.e., $\leq$ 71 scfm).

(continued)

PA1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.2 Primary Containment Air Locks

(personnel access hatch and emergency escape hatch)

BASES

BACKGROUND

Two double door primary containment air locks have been built into the primary containment to provide personnel access to the primary containment and to provide primary containment isolation during the process of personnel entry and exit. The air locks are designed to withstand the same loads, temperatures, and peak design internal and external pressures as the primary containment (Ref. 1). As part of the primary containment, the air lock limits the release of radioactive material to the environment during normal unit operation and through a range of transients and accidents up to and including postulated Design Basis Accidents (DBAs).

PA2 5 Plant PA1

PA1 Each of the personnel access hatch doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in primary containment internal pressure results in increased sealing force on each door).

Each air lock door has been designed and tested to certify its ability to withstand pressure in excess of the maximum expected pressure following a DBA in primary containment. ~~Each of the doors has inflatable seals that are maintained > [60] psig by the seal air flask and pneumatic system, which is maintained at a pressure ≥ [90] psig. Each door has two seals to ensure they are single failure proof in maintaining the leak tight boundary of primary containment.~~

DB3

Each air lock is nominally a right circular cylinder, 10 ft 2 inches in diameter, with doors at each end that are interlocked to prevent simultaneous opening. The air locks are provided with limit switches on both doors in each air lock that provide control room indication of door position. ~~[Additionally, control room indication is provided to alert the operator whenever an air lock interlock mechanism is defeated.]~~ During periods when primary containment is not required to be OPERABLE, the air lock interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions, as allowed by this LCO, the primary containment may be accessed through the air lock when the door interlock mechanism has failed, by manually performing the interlock function.

DB2

The primary containment air locks form part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining the primary containment leakage rate to within limits in the event of a

PA2 the

(continued)

REVISION I

BASES

BACKGROUND (continued)

PA1

DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the ~~unit~~ Plant safety analysis.

PA2

APPLICABLE SAFETY ANALYSES

The postulated DBA that ~~postulates~~ results in the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment ~~leakage~~. The for the primary containment is ~~designed with a~~ maximum allowable leakage rate (L_a) of ~~0.437%~~ by weight of the containment ~~and drywell~~ air per 24 hours at the ~~calculated~~ maximum peak containment pressure (P_a) of ~~11.5~~ psig design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with (Primary Containment Leakage Rate Testing Program) the air locks.

CLB2 1.5

PA2

CLB2 4.5

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

Primary containment air locks satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii) (Ref. 2).

X1

1.5

LCO

TA2

As part of the primary containment pressure boundary, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment air locks are required to be OPERABLE. For each air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be open at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single door in each air lock is

(continued)

REVISION ~~E~~ J

RAI 3.6.1.2-3

BASES

ACTIONS (continued)

D.1 and D.2

If the inoperable primary containment air lock cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

TA2

SR 3.6.1.2.1

X2

approved in License Amendment 97 (Ref. 3)

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of ~~10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions~~. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were ~~established during initial air lock and primary containment OPERABILITY testing~~. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by ~~10 CFR 50, Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~

the Primary Containment Leakage Rate Testing Program

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR, requiring the results to be evaluated against the acceptance criteria ~~of SR 3.6.1.1.1.~~ This ensures that air lock leakage is properly accounted for in determining the ~~overall~~ primary containment leakage rate.

PA2

(Primary Containment Leakage Rate Testing Program)

TA2

combined Type B and C

which is applicable to

(continued)

REVISION ~~E~~ J

Subsequently, License Amendment 261 (Ref. 4) allowed an increased overall air lock leakage rate (i.e., Amendment 261 increased the value of La; therefore, the overall air lock leakage rate limit value that corresponds to 0.05 La increased).

J

BASES

SURVEILLANCE REQUIREMENTS

DB4

SR 3.6.1.2.4 (continued)

~~plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the [18] month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

REFERENCES PA1 U

1. ~~F~~USAR, Section ~~[3.8]~~ 5.2. DB5
2. ~~10 CFR 50, Appendix J~~ 10 CFR 50.36 (c)(2)(ii). X1
3. ~~FSAR, Table [6.2-13].~~ NRC letter dated November 21, 1985, Issuance of Amendment 97 to the Facility Operating License DPR-59 for James A. FitzPatrick Nuclear Power Plant. X2
4. NRC letter dated April 14, 2000, Issuance of Amendment 261 to the Facility Operating License DPR-59 for James A. FitzPatrick Nuclear Power Plant. J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1434, REVISION 1
ITS BASES: 3.6.1.2 - PRIMARY CONTAINMENT AIR LOCKS

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1434, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

X2 The bracketed method to establish the air lock leakage limits in SR 3.6.1.2.1 has been revised to be consistent with plant specific method. References have been added as a result of this modification. J

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	12 hours 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 criteria applicable to SR 3.6.1.1.1. --------------- Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.2.2	Verify only one door in the primary containment air lock can be opened at a time.	24 months

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.2 Primary Containment Air Locks

BASES

BACKGROUND

Two double door primary containment air locks (personnel access hatch and emergency escape hatch) have been built into the primary containment to provide personnel access to the drywell and to provide primary containment isolation during the process of personnel entry and exit. The air locks are designed to withstand the same loads, temperatures, and peak design internal and external pressures as the primary containment (Ref. 1). As part of the primary containment, the air locks limit the release of radioactive material to the environment during normal plant operation and through a range of transients and accidents up to and including postulated Design Basis Accidents (DBAs).

Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a DBA in primary containment. Each of the personnel access hatch doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in primary containment internal pressure results in increased sealing force on each door).

Each air lock is nominally a right circular cylinder, with doors at each end that are interlocked to prevent simultaneous opening. The air locks are provided with limit switches on both doors in each airlock that provide control room indication of door position. During periods when primary containment is not required to be OPERABLE, the air lock interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions, as allowed by this LCO, the primary containment may be accessed through the air lock when the interlock mechanism has failed, by manually performing the interlock function.

The primary containment air locks form part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining the primary containment leakage rate to within limits in the event of a

(continued)

BASES

BACKGROUND (continued)	DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the plant safety analysis.
APPLICABLE SAFETY ANALYSES	The postulated DBA that results in the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The maximum allowable leakage rate (L_a) for the primary containment is 1.5% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 45 psig (Primary Containment Leakage Rate Testing Program). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks. Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment. The primary containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).
LCO	As part of the primary containment pressure boundary, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event. The primary containment air locks are required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be opened at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed

(continued)

BASES

LCO (continued)

when the air lock is not being used for normal entry into or exit from primary containment.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the primary containment air locks are not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

ACTIONS

The ACTIONS are modified by Note 1, which allows entry and exit to perform repairs of the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door, it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the primary containment boundary is not intact (during access through the OPERABLE outer door). The allowance to open the OPERABLE door, even if it means the primary containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the primary containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

Note 2 has been included to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are modified by a third Note, which ensures appropriate remedial measures are taken when necessary, if

(continued)

BASES

ACTIONS (continued)

air lock leakage results in exceeding overall containment leakage rate acceptance criteria. Pursuant to LCO 3.0.6, ACTIONS are not required even if primary containment leakage is exceeding L_a. Therefore, the Note is added to require ACTIONS for LCO 3.6.1.1, "Primary Containment," to be taken in this event.

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

With one primary containment air lock door inoperable in one or more primary containment air locks, the OPERABLE door in each affected air lock must be verified closed (Required Action A.1). This ensures that a leak tight primary containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, which requires that primary containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 ensures that the affected air lock penetration has been isolated by the use of a locked closed OPERABLE air lock door. This ensures that an acceptable primary containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate given the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas or areas with limited access due to inerting and allows these doors to be verified locked closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

(continued)

BASES

ACTIONS

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

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from the primary containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas or areas with limited access due to inerting and allows these doors to be verified locked closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

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

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate containment overall leakage rates using current air lock leakage test results. An evaluation is acceptable since it is overly conservative to immediately declare the primary containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed) primary containment remains OPERABLE, yet only 1 hour (according to LCO 3.6.1.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected primary containment air locks must be verified closed. This Required Action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1.1, which require that primary containment be restored to OPERABLE status within 1 hour.

(continued)

BASES

ACTIONS

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

Additionally, the air lock must be restored to OPERABLE status within 24 hours (Required Action C.3). The 24 hour Completion Time is reasonable for restoring an inoperable air lock to OPERABLE status considering that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable primary containment air lock cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. 1J

SURVEILLANCE REQUIREMENTS

SR 3.6.1.2.1

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were approved in License Amendment 97 (Ref. 3). Subsequently, License Amendment 261 (Ref. 4) allowed an increased overall air lock leakage rate (i.e., Amendment 261 increased the value of L_a; therefore, the overall air lock leakage rate limit value that corresponds to 0.05 L_a increased). The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program. 1J J

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR, requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1.1 (Primary Containment Leakage Rate Testing Program). This ensures that air lock leakage is properly accounted for in determining the combined Type B and C primary containment leakage rate. 1J

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2.2

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident primary containment pressure (Ref. 1), closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when primary containment air lock is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. 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 loss of primary containment OPERABILITY if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the air lock.

REFERENCES

1. UFSAR, Section 5.2.

2. 10 CFR 50.36(c)(2)(ii).

3. NRC letter dated November 21, 1985, Issuance of Amendment 97 to the Facility Operating License DPR-59 for James A. FitzPatrick Nuclear Power Plant.

4. NRC letter dated April 21, 2000, Issuance of Amendment 261 to the Facility Operating License DPR-59 for James A. FitzPatrick Nuclear Power Plant.

Specification 3.6.1.3 A1

3.7 (cont'd)

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

see ITS 3.6.2.1

[LCO 3.6.1.3]

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F, and fuel is in the reactor vessel, except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE A8

add second Applicability M1

see ITS: 3.10.8

[Applicability MODES 1, 2 and 3] M1

add proposed ACTION D for MSIV leakage not within limits L9

add proposed ACTION E for any LPCI or CS AOV PCIV leakage not within limits L10

4.7 (cont'd)

L13 In accordance with the primary Containment Leakage Rate Testing Program

see ITS: 3.6.1.1

2. a. Perform required visual examination and leakage rate testing of the Primary Containment in accordance with the Primary Containment Leakage Rate Testing Program.

[SR 3.6.1.3.10]

b. Demonstrate leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig. The testing frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

L12

[SR 3.6.1.3.11]

c. ~~Once per 24 months~~, demonstrate the leakage rate of 10AOV-68A,B for the Low Pressure Coolant Injection system and 14AOV-13A,B for the Core Spray system ~~to be less than 1 scfm per valve when pneumatically tested at ≥ 45 psig at ambient temperature, or less than 10 gpm per valve if hydrostatically tested at ≥ 1,035 psig at ambient temperature.~~ J

10 M6

within limits. LA1

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Revision J

ADMINISTRATIVE CHANGES

A3 CTS 3.7.D.2 requirement, to maintain at least one isolation valve operable in each affected penetration that is open, is being deleted. Proposed ITS 3.6.1.3 Condition A Note has been provided to restrict the applicability to penetrations with two or more PCIVs, where a second valve is available. This Note is consistent with the Notes provided in the new proposed ITS 3.6.1.3 Condition B (L3) for two or more valves inoperable in a penetration with two or more PCIVs, and ITS 3.6.1.3 Condition C (L4) for penetrations with only one PCIV. The addition of this Note identifying the applicable configuration, in conjunction with the separate and specific requirements provided in the proposed Conditions, is consistent with the format of NUREG-1433, Revision 1. Since there is no change in any technical requirements, this change is considered administrative.

A4 The requirement in CTS 3.7.D.2.a, to "restore the inoperable valve(s) to operable status within 4 hours," has been deleted since this is always an option. Since the time requirements on the alternative actions (CTS 3.7.D.2.b and 3.7.D.2.c are identical this change is considered administrative.

A5 The requirement to record the results in CTS 4.7.D.2 (ITS 3.6.1.3 Required Actions A.2 and C.2) is proposed to be deleted. This requirement duplicates the requirements of 10 CFR 50 Appendix B, Section XVII (Quality Assurance Records) to maintain records of activities affecting quality, including the results of tests/verifications. Compliance with 10 CFR 50 Appendix B is required by the JAFNPP Operating License. The details of the regulations within the Technical Specifications are repetitious and unnecessary. Therefore, retaining the requirement to perform the associated verifications and eliminating the details from Technical Specifications that are found in 10 CFR 50 Appendix B is considered a presentation preference, which is administrative.

A6 Not Used.

A7 Not Used.

A8 CTS 3.7.A.2 (3.7.D.1) requirement for primary containment isolation valves (PCIVs) to be Operable, has been revised. Proposed ITS LCO 3.6.1.3 provides an exception for reactor building-to-suppression

TECHNICAL CHANGES - MORE RESTRICTIVE

M4 (continued)

- SR 3.6.1.3.3, verify (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) each PCIV manual isolation valve, or blind flange that is located inside of primary containment and not locked, sealed or otherwise secured and is required to be closed during accident conditions is closed.
- SR 3.6.1.3.4, verify (each 31 days) continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.
- SR 3.6.1.3.9, remove and test (each 24 months on a STAGGERED TEST BASIS) the explosive squib from each shear isolation valve of the Tip System.

These SRs provide the means of ensuring the PCIVs are OPERABLE and able to perform their safety function which is to provide primary containment isolation. The addition of new Surveillance Requirements, imposes additional operational requirements, and constitutes a more restrictive change. This change is not considered to result in any reduction to safety.

M5 CTS 3.7.D.3 (CTS 3.7.A.8) requirement, that the reactor to be in the cold condition within 24 hours if the requirements of CTS 3.7.D.1 or 3.7.D.2 (CTS 3.7.A.1 through 3.7.A.5) associated with inoperable PCIVs cannot be met, is being changed. Allowances have been added to the current requirements to allow additional time to restore inoperable PCIVs, however these changes are addressed in L1, L3, L4, L9, and L10. ITS 3.6.1.3 Required Action F.1 requires the plant to be in MODE 3 in 12 hours if the Required Action and associated Completion Times for Condition A, B, C, D, or E are not met in MODE 1, 2, or 3. In addition, ITS 3.6.1.3 Required Action F.2 places the plant in MODE 4 in 36 hours (L7). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach the required plant condition from full power conditions in an orderly manner without challenging plant systems and is consistent with the requirements of NUREG-1433, Revision 1. Since, this change imposes additional operational and time requirements it is considered to be more restrictive. This change is not considered to result in any reduction to safety.

M6 CTS 4.7.A.2.c requires the LPCI and Core Spray Systems air operated testable check valves to be leak tested, with the limit ≤ 11 scfm per valve when pneumatically tested at ≥ 45 psig at ambient temperature. The pneumatic test limit is being decreased to ≤ 10 scfm as shown in the Bases for SR 3.6.1.3.11 (DOC LA1 describes moving the limit to the

TECHNICAL CHANGES - MORE RESTRICTIVE

M6 (continued)

Bases). The new limit is based on the NRC Safety Evaluation Report to Technical Specification Amendment No. 40, dated November 9, 1978, which specifies the pneumatic test limit as ≤ 10 scfm. 5

M7 CTS 4.7.B.4 requirement, that 27MOV-120 (12 inch, full-flow valve) be verified closed when containment integrity is established, and then once per month, is being revised. ITS SR 3.6.1.3.1, requires verification that each 20 and 24 inch primary containment purge and vent valve is closed every 31 days. Since the purge and vent valves are the actual primary containment isolation valves (PCIVs) associated with these penetrations, this change is appropriate. Since CTS 3.7.B.4 allows inerting and de-inerting operations only with valve 27MOV-121 (6 inch, low flow valve) it is understood that the primary containment purge and vent valves must be opened for these operations. Therefore, a Note has been added to proposed SR 3.6.1.3.1 which allows these operations to occur as long as the full-flow line (27MOV-120) is closed and one or more Standby Gas Treatment (SGT) System reactor building suction valves are open. This provides protection of the SGT filter trains from over pressure concerns. This change is considered more restrictive since the primary containment vent and purge valves are required to be closed when these operations are not underway. This is consistent with current practice and in accordance with the UFSAR safety analyses. This assures that the requirements of the LOCA are met and ensures these valves are opened for a valid reason. This change is not considered to result in any reduction to safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 Requirements in CTS 4.7.A.2.c concerning the leakage limit and test pressure for LPCI/CS air operated testable check valves are proposed to be relocated to the Bases. The leakage limits and test pressure (as modified by DOC M6) are not necessary for ensuring the test is performed. The requirements of ITS 3.6.1.3 and SR 3.6.1.3.11 are adequate to ensure the OPERABILITY of these valves and that they are tested properly. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS. 5

LA2 Not Used.

LA3 Not Used.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA4 Not Used.

LA5 Details in CTS 1.0.M.3, definition of Primary Containment Integrity (OPERABILITY), concerning automatic containment isolation valves (a de-activated valve in the isolated position ensures containment integrity) is being relocated to the Bases. The details for valve OPERABILITY are not necessary to ensure the Primary Containment Isolation Valves are OPERABLE. The requirements of ITS 3.6.1.3 which require the PCIVs to be OPERABLE and the definition of OPERABILITY suffice. ITS LCO 3.6.1.3 Bases clearly states that an automatic isolation valve is OPERABLE if de-activated and secured in the closed position. As such, these details are not required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS.

LA6 Design details in CTS 3.7.D.1, which provide the containment vent and purge Valve Numbers and Maximum Opening Angle limitations, are to be relocated to the UFSAR. These design details are not necessary to be included in the Technical Specifications to ensure the OPERABILITY of these Primary Containment Isolation Valves. The requirements of ITS 3.6.1.3 are adequate to ensure the PCIVs are maintained OPERABLE. The design details are not required to be in the ITS to provide adequate protection of public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 4.7.D.1.a and CTS Table 4.2-1 Note 7, for actuation testing of PCIVs, stipulates a simulated automatic actuation test shall be performed. ITS SR 3.6.1.3.7 allows for use of an actual isolation signal, in addition to the simulated automatic initiation signal, for verifying that each PCIV actuates on an automatic initiation signal. This allows satisfactory actual automatic system initiations to be used to fulfill the Surveillance Requirements. Operability is adequately demonstrated in either case since the PCIVs cannot discriminate between

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 (continued)

"actual" or "simulated" signals. This change, to allow the use of actual automatic initiation signals, provides increased latitude for operations to complete the Surveillance Requirement and is therefore considered to be less restrictive.

L2 Not Used.

L3 CTS 3.7.D does not provide specific ACTIONS for those penetrations with two or more inoperable PCIVs unless the penetration is closed and no operable valves are required (CTS 3.7.D.2). ITS 3.6.1.3 ACTION B, to isolate the affected penetration flow path within 1 hour when one or more penetration flow paths exist with two or more PCIVs inoperable, for reasons other than Conditions D and E, is being added. Currently entry into CTS 3.7.D.3 is required and the plant must be in cold condition in 24 hours. The additional 1 hour allowed to isolate the affected penetration flow path provides a period of time to correct the problem commensurate with the importance of maintaining primary containment OPERABILITY during MODES 1, 2, and 3. Additionally, the one hour period ensures that the probability of an accident (requiring primary containment OPERABILITY) occurring during periods where primary containment is inoperable is minimized. This change, to allow 1 hour to isolate the affected penetration, provides relief for the current operational requirements to commence a plant shutdown, and therefore, is considered to be less restrictive.

L4 CTS 3.7.D does not provide specific ACTIONS for those penetration flow paths with one PCIV. Currently entry into CTS 3.7.D.3 is required and the plant must be in cold condition in 24 hours. ITS 3.6.1.3 ACTION C requires the affected penetration flow path to be isolated within 4 hours except for EFCVs and penetrations with a closed system and within 72 hours for EFCVs and penetrations with a closed system. The 4 hour Completion Time is acceptable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY. The 72 hour Completion Time is acceptable since the associated penetrations are part of a closed system which will act as a barrier or are of small diameter (EFCV penetrations). During the allowed time, a limiting event would still be assumed to be within the bounds of the safety analysis. Allowing this extended time potentially avoiding a plant transient caused by the immediate forced shutdown, is reasonable based on the low probability of an event, and does not represent a significant decrease in safety. In addition, to ensure the affected penetration are isolated on a periodic basis, Required Action C.2 has been added. Required Action C.2 will require the verification that each affected penetration flow path is isolated

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

once per 31 days. The 31 day Frequency is acceptable since the devices are operated under administrative controls and the probability of misalignment is low.

L5 CTS 3.7.D.2.b Completion Time of 4 hours, to isolate each affected penetration has been extended for certain penetrations (ITS 3.6.1.3 ACTION A). For penetrations with two or more PCIVs, proposed Required Action A.1 allows 8 hours for main steam line penetrations and 4 hours for other penetrations. During the allowed time, the limiting event would still be assumed to be within the bounds of the safety analysis since a second valve is available for isolation or in the case of EFCV penetrations, no credit is taken for isolation since the installed orifice will limit the leakage to within limits. This change is acceptable since the 8 hour Completion Time for MSIVs allows time to repair or reduce power to isolate the affected penetration. Allowing this additional time potentially avoids a plant transient caused by a reduction in power to close the MSIVs.

L6 A new method of isolating penetrations is proposed to be added to CTS 3.7.D.2.c when one or more penetration flow paths with one PCIV is inoperable (except for when MSIV or hydrostatically tested valve leakage is not within limits). ITS 3.6.1.3 Required Action A.1 allows the penetration to be isolated by a check valve with flow through the valve secured. This is acceptable for penetrations with only one PCIV inoperable because the other PCIV remains Operable, the likelihood of an event occurring in which a containment isolation is required is remote, the penetration is isolated by a check valve, and the remaining Operable PCIV not being able to also isolate the penetration is remote. This description has also been added to the Bases to describe a passive PCIV.

L7 CTS 3.7.D.3 (CTS 3.7.A.8) requirement, that the reactor be in the cold condition within 24 hours if the requirements of CTS 3.7.D.1 or 3.7.D.2 (3.7.A.1 through 3.7.A.5) with respect to PCIVs cannot be met, is being relaxed. Allowances have been added to the current requirements to allow additional time to restore inoperable PCIVs, however these changes are addressed in L1, L3, L4, L9, and L10. Proposed ITS 3.6.1.3 Required Action F.2 allows the plant 36 hours to reach COLD SHUTDOWN (MODE 4) if the Required Action and Completion Time of Condition A, B, C, D, or E cannot be met in MODE 1, 2, or 3. However, ITS 3.6.1.3 Required

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L9 (continued)

isolating the penetration, the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown, and the relative importance of leakage to the overall containment function. This change is acceptable since the closure of one MSIV in each penetration flow path will ensure the consequences of a design basis accident will be bounded by the USFAR analysis.

L10 A new ACTION has been added to CTS 3.7.A.2 (ITS 3.6.1.3 ACTION E) which will allow 72 hours to restore leakage rate to within limit for one or more air operated testable check valves associated with the Low Pressure Coolant Injection and Core Spray Systems injection penetrations. The additional 72 hours to restore leakage within the limit provides a period of time to correct the problem commensurate with the importance of maintaining primary containment Operability in MODES 1, 2 and 3. The associated penetrations are normally isolated during plant operations by a motor operated PCIV. In addition, there is an additional motor operated valve (which is hydrostatically leak tested under the IST program) available to isolate the penetration. Therefore, excessive leakage will be minimized by this closed motor operated PCIV and therefore ALARA concerns in the reactor building will be minimized. In the event of a pipe rupture outside of containment gross leakage is limited by the air operated testable check valve inside containment, however if it is inoperable the motor operated PCIV will also minimize the leakage. The reactor building includes radiation monitors which will provide audible and visual alarms to the control room. The Keep Full low level alarms and the reactor building floor drain sump high level alarms are available to indicate excessive primary coolant leakage. Therefore, since isolation methods exists to limit the leakage and since the plant is instrumented with diverse methods to detect leaks within the reactor building this 72 hour allowance is acceptable. This time is consistent with the Completion Times for other penetration flow paths with two or more PCIVs (one PCIV inoperable for reasons other than leakage) as indicated in ITS 3.6.1.3 Action A. 1J

L11 CTS 4.7.D.2 Surveillance Requirement, to verify (each 31 days) that a penetration flow path with an inoperable PCIV is isolated, is being supplemented. ITS 3.6.1.3 Required Actions A.2 and C.2 include two Notes. Note 1 allows isolation devices in high radiation areas to be verified by use of administrative means. This allowance is considered acceptable since access to these areas is typically restricted, and therefore the probability of misalignment once they have been verified to be in the proper position is low and the allowance is also consistent

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES (PCIVs)

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

The Licensee has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change would allow either 4 hours or 72 hours, depending upon the type of penetration, to isolate a primary containment penetration in those penetrations with one PCIV and allow operation to continue after the penetration flow path is isolated. Primary containment isolation is not an initiator of any previously analyzed accident. Therefore, this change does not increase the probability of such accidents. During the 4 hour or 72 hour allowed time, a limiting event would still be assumed to be within the bounds of the safety analysis since the isolation capability is still maintained by the closed system. Allowing this extended time to potentially avoid a plant transient caused by the immediate forced shutdown, is reasonable based on the low probability of an event, and does not represent a significant decrease in safety. The consequences of an event that may occur during the extended Completion Time would not be any different than during the currently allowed Completion Time. Therefore, this change does not significantly increase the consequences of any previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Further, since the change impacts only the Completion Time for the penetration isolation and does not result in any change in the response of the equipment to an accident, the change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

3. Does this change involve a significant reduction in a margin of safety?

   This change impacts only the Completion Time for inoperable valves that provide containment isolation. The methodology and limits of the accident analysis are not affected, nor is the containment response affected. Therefore, the change does not involve a significant

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

[CTS 3.7.D.1]
[CTS 3.7.A.2] LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.
[1-O.M]

APPLICABILITY: MODES 1, 2, and 3,
When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."
[3.7.D.1]
[3.7.A.2] [M1]

ACTIONS

CLB1

-----NOTES-----

[3.7.D.2.b] [1.O.M.1]
1. Penetration flow paths ~~[except for purge valve penetration flow paths]~~ may be unisolated intermittently under administrative controls.

[A2]
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 ~~in MODES 1, 2, and 3~~. PA1

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----NOTE----- Only applicable to penetration flow paths with two (TAS) or more PCIVs. ----- [3.7.D.2] One or more penetration flow paths with one PCIV inoperable ~~[except for purge valve leakage not within limit]~~ for reasons other than Conditions D and E. X1 X8	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. AND	4 hours except for main steam line AND 8 hours for main steam line 1 (I)

(continued)

REVISION ~~E~~ J

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
[L3] B. ---------NOTE--------- Only applicable to penetration flow paths with two or more PCIVs. --------------------- TA5 One or more penetration flow paths with two PCIVs inoperable ~~except for purge valve leakage not within limit~~. for reasons other than Conditions D and E	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour TA5
[L4] C. ---------NOTE--------- Only applicable to penetration flow paths with only one PCIV. --------------------- TA5 One or more penetration flow paths with one PCIV inoperable for reasons other than Conditions D and E	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND 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. [L11] TA6 [4.7.D.2] [L8] --------------------- Verify the affected penetration flow path is isolated.	~~4~~ hours except for excess flow check valves (EFCVs) and penetrations with a closed system TA3 AND 72 ~~12~~ hours ~~for EFCVs~~ and penetrations with a closed system TA6 Once per 31 days
[L9] D. ~~Secondary containment bypass~~ leakage rate ~~not within~~ limit. One or more penetration flow paths with one or more MSIVs not within X1 TA5	D.1 Restore leakage rate to within limit.	~~4~~ 8 hours

(continued)

REVISION ~~E~~ J

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.~~8~~ ①	------------------NOTES------------------ ~~1. Only required to be met in MODES 1, 2, and 3.~~ ~~2~~. Not required to be met when the ~~[18]~~ 20 inch and 24 inch primary containment vent and 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 the full flow line to Standby Gas Treatment (SGT) System is closed and one or more SGT System reactor building suction valves are open. --- Verify each ~~[18]~~ 20 inch and 24 inch primary containment vent and purge valve is closed.	31 days
SR 3.6.1.3.~~9~~ 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

CLB1 PA1 PA3 X3 J CLB4 J J DB1 PA3 CLB1 TA1

[M7] [3.7.B.4] [1.0.M.4] [1.0.M.1] [M4]

(continued)

Revision J

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.1.3. CLB1 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 manual isolation valve and blind flange that is located inside primary containment and is required to be closed during accident conditions is closed. and not locked, sealed or otherwise secured TA1	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 CLB1	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3. CLB1 6	Verify J TA2 the isolation time of each power operated ~~and each~~ automatic PCIV, except for MSIVs, is within limits.	[In accordance with the Inservice Testing Program ~~or 92 days~~] CLB5

[1.0.M.1]
[1.0.M.4]
[M4]
[M4]
[4.7.D.1.a]
[4.7.D.1.c]
J

(continued)

Revision J

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB9 ITS 3.6.1.3 has been revised to reflect the current licensing requirements of JAFNPP, that since no separate secondary containment bypass leakage is considered with respect to the primary containment leakage, no specific leakage rates or Surveillance Requirements exist in the CTS 3/4.7. The bracketed ISTS 3.6.1.3 Action D reference to secondary containment bypass leakage and the bracket SR 3.6.1.3.12 to verify secondary containment bypass leakage path limits are not applicable and have been deleted. Subsequent Surveillance Requirements have been renumbered as applicable.

CLB10 Not Used.

CLB11 ITS SR 3.6.1.3.11 (ISTS SR 3.6.1.3.14) has been revised to reflect the current licensing requirement of JAFNPP, CTS 4.7.A.2.c (as modified by DOC M6), to determine the leakage rate of each air operated testable check valve associated with the LPCI and CS System vessel injection penetrations.

CLB12 ITS SR 3.6.1.3.7 has been revised to reflect the requirements at JAFNPP that the Frequency for verifying each automatic PCIV actuates to the isolation position on an actual (L1) or simulated isolation signal is 24 months (A9) consistent with CTS Table 4.2-1, Primary Containment Isolation Instrumentation Test and Calibration Requirements.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The words "in MODES 1, 2, and 3" have been deleted from ITS 3.6.1.3 ACTIONS Note 4 since there are no PCIV leakage tests required in MODES other than MODES 1, 2, and 3 for JAFNPP (i.e., there are no PCIVs required to be OPERABLE in MODES other than MODES 1, 2, and 3 that have specific leakage limits). In addition, ITS SR 3.6.1.3.1, Note 1 and SR 3.6.1.3.11 Note 1, have been deleted for the same reason. The subsequent Notes have been renumbered, as applicable.

PA2 Editorial changes have been made to enhance clarity.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES (PCIVs)

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA3 The plant specific terminology has been included.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 ITS 3.6.1.3 has been revised to reflect specific differences based on the JAFNPP design of the vent and purge system. The vent and purge valves at JAFNPP are of two sizes, 20 inch and 24 inch.

DB2 Not used.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 45, Revision 2, have been incorporated into the revised Improved Technical Specifications.

TA2 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 46, Revision 1, have been incorporated into the revised Improved Technical Specifications.

TA3 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 30, Revision 3, have been incorporated into the revised Improved Technical Specification.

TA4 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 52, Revision 3, have been incorporated into the revised Improved Technical Specifications.

TA5 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 207, Revision 5, have been incorporated into the revised Improved Technical Specifications.

TA6 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 269, Revision 2, have been incorporated into the revised Improved Technical Specification.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.3 Primary Containment Isolation Valves (PCIVs)

BASES

BACKGROUND

The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) to within limits. Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The OPERABILITY requirements for PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Therefore, the OPERABILITY requirements provide assurance that primary containment function assumed in the safety analyses will be maintained. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. ~~Two~~ barriers ~~in series~~ are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system.

PA1
and
One or more
J
PA1
When two or more barriers are provided
5

The reactor building-to-suppression chamber vacuum breakers serve a dual function, one of which is primary containment isolation. However, since the other safety function of the vacuum breakers would not be available if the normal PCIV actions were taken, the PCIV OPERABILITY requirements are not applicable to the reactor building-to-suppression chamber vacuum breakers valves. Similar surveillance requirements in the LCO for reactor building-to-suppression chamber vacuum breakers provide assurance that the isolation capability is available without conflicting with the vacuum relief function.

(continued)

PA2 DB1

INSERT BKGD-1

BASES

BACKGROUND (continued)

The primary containment ~~purge lines are [18] inches in diameter; vent lines are [18] inches in diameter. The [18] inch primary containment purge valves~~ are normally maintained closed in MODES 1, 2, and 3 to ensure the primary containment boundary is maintained. ~~The isolation valves on the [18] inch vent lines have [2] inch bypass lines around them for use during normal reactor operation. Two additional redundant excess flow isolation dampers are provided on the vent line upstream of the Standby Gas Treatment (SGT) System filter trains. These isolation dampers, together with the PCIVs, will prevent high pressure from reaching the SGT System filter trains in the unlikely event of a loss of coolant accident (LOCA) during venting. Closure of the excess flow isolation dampers will not prevent the SGT System from performing its design function (that is, to maintain a negative pressure in the secondary containment). To ensure that a vent path is available, a [2] inch bypass line is provided around the dampers.~~

DB1 CLB4

INSERT BKGD-2

these valves

DB1

APPLICABLE SAFETY ANALYSES

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

PA1

The DBAs that result in a release of radioactive material ~~within primary containment~~ are a LOCA, and a main steam line break (MSLB). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or close within the required isolation times following event initiation. This ensures that potential paths to the environment through PCIVs (including primary containment purge valves) are minimized. Of the events analyzed in Reference 1, the MSLB is the most limiting event due to radiological consequences. The closure time of the main steam isolation valves (MSIVs) is a significant variable from a radiological standpoint. The MSIVs are required to close within 3 to 5 seconds since the 5 second closure time is assumed in the analysis. ~~The safety analyses assume that the purge valves were closed at event initiation.~~ Likewise, it is assumed that the primary

PA1
for which the consequences are mitigated by PCIVs

, control rod drop accident,

DB11

PA2
vent and

DB3
to control room personnel

PA2
, after signal generation,

e

consistent with or conservative to the times

DB4

MSLB DB4

CLB1

DB5

Refs. 3 and 4

DB4

(continued)

3 second closure time is assumed in the MSIV closure analysis (Ref. 2) and the

DB5

REVISION ~~E~~ J

DB 8 INSERT ASA-1

BASES

APPLICABLE SAFETY ANALYSES (continued)

containment is isolated such that release of fission products to the environment is controlled.

DB8

~~The DBA analysis assumes that within 60 seconds of the accident, isolation of the primary containment is complete and leakage is terminated, except for the maximum allowable leakage rate, L_a. The primary containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and PCIV stroke times.~~

DB2 plant PA2 vent and

The single failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the primary containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred.

CLB1

~~[The primary containment purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, and 3. In this case, the single failure criterion remains applicable to the primary containment purge valve due to failure in the control circuit associated with each valve. The primary containment purge valve design precludes a single failure from compromising the primary containment boundary as long as the system is operated in accordance with this LCO.]~~

PCIVs satisfy Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36 (c)(2)(ii) (Ref. 7) X8 1J

LCO

PCIVs form a part of the primary containment boundary. The PCIV safety function is related to minimizing the loss of reactor coolant inventory and establishing the primary containment boundary during a DBA.

DB1 20 and 24 vent and PA2 X7

The power operated, automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. The [18] inch purge valves must be maintained ~~sealed~~ closed *or blocked to prevent full opening*. While the reactor building-to-suppression chamber vacuum breakers isolate primary containment penetrations, they are excluded from this Specification. Controls on their isolation function are adequately addressed in LCO

(continued)

REVISION J

DB 8

INSERT ASA-1

does not assume a specific closure time for primary containment isolation valves (PCIVs). The analysis assumes that the leakage from the primary containment is 1.5 percent primary containment air weight per day (L_a) at pressure P_a throughout the accident. The bases for PCIV closure times, and the specified valve closure times, are specified in UFSAR Section 7.3.3.1 and UFSAR Table 7.3-1 (Refs. 5 and 6), respectively.

The associated stroke time of each automatic PCIV is included in the Inservice Testing (IST) Program

6 PA4

BASES

LCO (continued)

3.6.1.~~2~~, "Reactor Building-to-Suppression Chamber Vacuum Breakers." The valves covered by this LCO are listed ~~with their associated stroke times~~ in Reference ~~2~~. 8 X11 J

The normally closed PCIVs are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves and devices are those listed in Reference ~~2~~. 8 X11 J

~~Purge valves with resilient seals, secondary bypass valves,~~ CLB1 MSIVs, ~~and hydrostatically tested~~ CLB9 Low Pressure Coolant Injection (LPCI) and Core Spray (CS) System air operated testable Check CLB11 valves must meet additional leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment," as Type B or C testing.

This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the primary containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, most PCIVs are not required to be OPERABLE and the primary containment PA2 Vent and purge valves are not required to be PA2 normally ~~sealed~~ closed in MODES 4 and 5. Certain valves, however, are required to be OPERABLE to prevent inadvertent reactor vessel draindown. These valves are those whose associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation." (This does not include the valves that isolate the associated instrumentation.)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow path(s) ~~(except for purge valve flow path(s))~~ CLB1 to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous

(continued)

Revision J

BASES

CLB 1

ACTIONS (continued)

communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated. ~~Due to the size of the primary containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves is not allowed to be opened under administrative controls. A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.1.3.1.~~

A second Note has been added to provide clarification that, for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable PCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable PCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are modified by Notes 3 and 4. Note 3 ensures that appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable PCIV (e.g., an Emergency Core Cooling System subsystem is inoperable due to a failed open test return valve). Note 4 ensures appropriate remedial actions are taken when the primary containment leakage limits are exceeded. Pursuant to LCO 3.0.6, these actions are not required even when the associated LCO is not met. Therefore, Notes 3 and 4 are added to require the proper actions be taken.

J

leakage or inoperabilities due to MSIV, LPCI or CS System air operated testable check valve

X12 TAS

J

A.1 and A.2

CLB 1

With one or more penetration flow paths with one PCIV inoperable except for purge valve leakage not within limit, the affected penetration flow paths must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1,

(continued)

REVISION ~~E~~ J

BASES

ACTIONS

A.1 and A.2 (continued)

the device used to isolate the penetration should be the closest available valve to the primary containment. The Required Action must be completed within the 4 hour Completion Time (8 hours for main steam lines). The Completion Time of 4 hours is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. For main steam lines, an 8 hour Completion Time is allowed. The Completion Time of 8 hours for the main steam lines allows a period of time to restore the MSIVs to OPERABLE status given the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown.

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration flow path(s) must be verified to be isolated on a periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following an accident, and no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those devices outside containment and capable of potentially being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside primary containment" is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. For the devices inside primary containment, the time period specified "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" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the devices and other administrative controls ensuring that device misalignment is an unlikely possibility.

TA5 or more

Condition A is modified by a Note indicating that this Condition is only applicable to those penetration flow paths ~~with two~~ PCIVs. For penetration flow paths with one PCIV, Condition C provides the appropriate Required Actions. 1 (1)

two Notes. Note 1

TA6

Required Action A.2 is modified by ~~a Note that~~ applies to isolation devices located in high radiation areas, and

(continued)

REVISION ~~E~~ J

BASES

ACTIONS

A.1 and A.2 (continued)

TA6 Insert ACTIONS A.1

allows them to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment ~~of these devices~~, once they have been verified to be in the proper position, is low.

X9

B.1

TA5 X12 leakage or except for inoperabilities due to MSIV, LPCI or CS System air operated testable check valve leakage not within limits

PA1 component TA5 or more

With one or more penetration flow paths with two PCIVs inoperable, either the inoperable PCIVs must be restored to OPERABLE status or the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1.

J

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two PCIVs. For penetration flow paths with one PCIV, Condition C provides the appropriate Required Actions.

or more TA5

J

C.1 and C.2

With one or more penetration flow paths with one PCIV inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the [4] hour Completion Time. The Completion Time of [4] hours is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting primary containment OPERABILITY during

PA1 component

TA3 Insert C.1-A

72 for penetrations with a closed system

J

(continued)

Revision J

TA 6 INSERT ACTIONS A.1

Note 2 applies to the isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verfied closed by administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing of components is to ensure that these devices are not inadvertently repositioned. J

TA 3 INSERT C.1-A

The Completion Time of 4 hours is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. J

DB10

The closed system must meet the requirements of Reference 9.

TA7

TA3

72 hours for EFCVs is also

1 J

ACTIONS

C.1 and C.2 (continued)

MODES 1, 2, and 3. The Completion Time of 12 hours is reasonable considering the instrument and the small pipe diameter of penetration (hence, reliability) to act as a penetration isolation boundary and the small pipe diameter of the affected penetrations. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration must be verified to be isolated on a periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

PA1

INSERT C.1 and C.2

Condition C is modified by a Note indicating that this Condition is only applicable to penetration flow paths with only one PCIV. For penetration flow paths with two or more PCIVs, Conditions A and B provide the appropriate Required Actions.

TA5

J

Required Action C.2 is modified by ~~a Note that~~ two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is low.

TA6

INSERT Required Action C.2

D.1

PA1

any

CLB9

8

X1

With the secondary containment bypass leakage rate or MSIV leakage rate not within limit, the assumptions of the safety analysis may not be met. Therefore, the leakage must be restored to within limit within 4 hours. Restoration can be accomplished by isolating the penetration that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed

(continued)

REVISION J

X12 TA3

INSERT ACTION E

E.1

With one or more penetration flow paths with LPCI System or CS System air operated testable check valve leakage rate not within limits, the assumptions of the safety analysis may not be met. Therefore, the leakage must be restored to within limit within 72 hours. Restoration can be accomplished by isolating the penetration that caused the limit to be exceeded by use of one closed and de-activated automatic valve, or closed manual valve. When a penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 72 hour Completion Time is reasonable considering the time required to restore the leakage and the importance to maintain these penetrations availabe to perform the required function during a design basis accident. 1 J

CLB1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1 (continued)

containment occurs in these MODES, the purge valves may not be capable of closing before the pressure pulse affects systems downstream of the purge valves or the release of radioactive material will exceed limits prior to the closing of the purge valves. At other times when the purge valves are required to be capable of closing (e.g., during handling of irradiated fuel), pressurization concerns are not present and the purge valves are allowed to be open.

X3

X3

1 CLB1

vent and

PA2

SR 3.6.1.3.~~2~~ 1

This SR ensures that the primary containment vent and purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. ~~If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits.~~ ~~[The SR is also modified by a Note (Note 1), stating that primary containment purge valves are only required to be closed in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, the purge valves may not be capable of closing before the pressure pulse affects systems downstream of the purge valves, or the release of radioactive material will exceed limits prior to the purge valves closing. At other times when the purge valves are required to be capable of closing (e.g., during handling of irradiated fuel), pressurization concerns are not present and the purge valves are allowed to be open.]~~

PA3

The SR is modified by a Note (Note 2) stating that the SR is not required to be met when the vent and purge valves are open for the stated reasons. The Note states that these valves may be opened for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open, provided the full-flow 12 inch line (with valve 27MOV-120) to the SGT is closed and one or more SGT System suction valves are open. This will ensure there is no damage to the filters if a LOCA were to occur with the vent and purge valves open since excessive differential pressure is not expected with the full-flow 12 inch line closed and one or more SGT System suction valves open. The ~~[18]~~ 20 and 24 inch vent and purge valves are capable of closing ~~in the environment following~~ against the dynamic effects of a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other PCIV requirements discussed in SR 3.6.1.3.~~5~~ 2.

PA2

vent and

DB1

PA2

CLB4

1

reactor building

B 3.6-25

1

(continued)

CLB1

BASES

3

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.~~4~~ (continued)

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

PA1

PA1

SR 3.6.1.3.~~5~~ 4 CLB1

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

SR 3.6.1.3.~~6~~ 5 CLB1

TA2 J 15

Verifying the isolation time of each power operated ~~and each~~ automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.~~7~~ 6. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The ~~isolation time and~~ Frequency of this SR ~~are~~ is in accordance with the requirements of the Inservice Testing Program ~~or 92 days~~.

CLB1 6 CLB5 13

CLB1

~~[SR 3.6.1.3.7~~

~~For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, (Ref. 3), is required to ensure]~~

~~(continued)~~

Rasmusson J

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.~~8~~ 7 CLB1

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in ~~SR 3.3.6.1.7~~ LCO 3.3.6.1, "Primary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The ~~[18]~~ 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a ~~unit~~ plant outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience has shown that these components usually pass this Surveillance when performed at the ~~[18]~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

PA4 CLB12 PA2 CLB12

SR 3.6.1.3.~~10~~ 8 CLB1

This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) is OPERABLE by verifying that the valve ~~[reduces flow to ≤ 1 gph]~~ actuates to the isolation position on a simulated instrument line break]. This SR provides assurance that the instrumentation line EFCVs will perform so that ~~predicted radiological consequences~~ secondary containment will not be ~~exceeded~~ overpressurized during the postulated instrument line break ~~event evaluated in Reference [2]~~ (10). ~~The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program,

CLB8 CLB8 PA2 J J J

SR 3.6.1.3.~~11~~ 9 CLB1

The TIP shear isolation valves are actuated by explosive charges. An in-place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when

PA1

(continued)

BASES

PA2 CLB9 CLB10 DB7

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.12 10

The analyses in References 2 and 6 11 are based on leakage that is less more than the specified leakage rate. Leakage through each MSIV must be ≤ [11.5] 25 scfh when tested at ≥ [P] [28.8] psig. The MSIV leakage rate must be verified to be in accordance with the leakage test requirements of 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions. in accordance with the Primary Containment Leakage Rate Testing Program Note 1 is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2, and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions; thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

CLB10 TA4 PA3

CLB9 CLB11 CLB13

SR 3.6.1.3.13 11

Insert SR 3.6.1.3.11

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 2 are met. resulting radiation dose rate that would result if the reactor coolant were released to the reactor building at the specified limit will be small (Ref. 12) The combined leakage rates must be demonstrated in accordance with the leakage rate test Frequency of 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions; required by the Primary Containment Leakage Rate Testing Program thus SR 3.0.2 (which allows Frequency extensions) does not apply.

TA4 PA3

[This SR has been modified by a Note that states that these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3, since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage limits are not applicable in these other MODES or conditions.]

X7

The acceptance criteria for each air operated testable check valve associated with the LPCI and CS Systems vessel injection penetrations is < 1 gpm when hydrostatically tested at ≥ 1035 psig or < 10 scfm when pneumatically tested at ≥ 45 psig, at ambient temperature (Ref. 12).

SR 3.6.1.3.15

[Reviewer's Note: This SR is only required for those plants with purge valves with resilient seals allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices that are not permanently installed on the valves.]

(continued)

CLB11 TA4

Revision J

CLB11

INSERT SR 3.6.1.3.11

each air operated testable check valve associated with the LPCI and CS System vessel injection penetrations 1A

PA2

BASES

SURVEILLANCE REQUIREMENTS

X7

SR 3.6.1.3.15 (continued)

Verifying each [ ] inch primary containment purge valve is blocked to restrict opening to ≤ [50]% is required to ensure that the valves can close under DBA conditions within the times assumed in the analysis of References 2 and 6. [The SR is modified by a Note stating that this SR is only required to be met in MODES 1, 2, and 3.] If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The [18] month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

REFERENCES

1. FSAR, ~~Chapter [15]~~ Section 14.6. DB3
2. UFSAR, Section 14.5.2.3. DB4
3. FSAR, ~~Table [6.2-5]~~ Section 6.5.3.2. DB4
4. ~~10 CFR 50, Appendix J~~ UFSAR, Section 14.8.2.6.2 DB4 J
10. FSAR, Section ~~[6.2]~~ 14.3.2.5. TA7 J
11. FSAR, Section ~~[15.1.39]~~ 14.8.2.1.1 DB5 CLB12 J
12. NRC Letter to NYPA, November 9, 1978, NRC Safety Evaluation Supporting Amendment 40 to Facility Operating License No. DPR-59. J
8. Technical Requirements Manual. X11 J
7. 10 CFR 50.36 (c)(2)(ii). X8 J
9. UFSAR, Section 5.2.3.5. DB10 J
5. UFSAR, Section 7.3.3.1. J
6. UFSAR, Table 7.3-1. J

Revision 5

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Editorial changes have been made for enhanced clarity or to correct a grammatical/typographical error.

PA2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific system/structure/component nomenclature, equipment identification or description.

PA3 The information for ITS SR 3.6.1.3.1 Note 1, SR 3.6.1.3.10 Note 1, and SR 3.6.1.3.11 Note 1, has been deleted, since there are no PCIVs required to be OPERABLE in MODES other than MODES 1, 2, and 3 that have specific leakage limits for JAFNPP. Subsequent Notes are renumbered as applicable.

PA4 The correct LCO number has been provided.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 ITS 3.6.1.3 has been revised to reflect specific differences based on the JAFNPP design of the vent and purge system. The vent and purge valves at JAFNPP are of two sizes 20 inch and 24 inch.

DB2 ITS 3.6.1.3 APPLICABLE SAFETY ANALYSES has been revised to reflect specific differences based on the JAFNPP design of the vent and purge system. The brackets have been removed and the information retained, since the JAFNPP two valve configuration for purge and vent lines is consistent with meeting the single failure criterion.

DB3 ITS 3.6.1.3 has been revised to reflect the specific JAFNPP reference requirements of UFSAR, Section 14.6, Analysis of Design Basis Accidents.

DB4 ITS 3.6.1.3 has been revised to reflect the specific JAFNPP reference requirements of UFSAR, Sections 6.5.3.2 and 14.8.2.1.2, Steam Line Breaks.

DB5 ITS 3.6.1.3 has been revised to reflect the specific JAFNPP reference requirements of UFSAR, Section 14.5.2.3, Main Steam Line Isolation Valve Closure.

DB6 Not used.

DB7 ITS 3.6.1.3 has been revised to reflect the specific JAFNPP reference requirements of UFSAR, Section 14.8.2.1.1, Loss of Coolant Accident.

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 for reasons other than Conditions D and E.	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 <u>AND</u> 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
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 for reasons other than Conditions D and E.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------NOTE--------- Only applicable to penetration flow paths with only one PCIV. --------------------- One or more penetration flow paths with one PCIV inoperable for reasons other than Conditions D and E.	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>		
	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)

J

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more penetration flow paths with one or more MSIVs not within leakage rate limit.	D.1	Restore leakage rate to within limit.	8 hours
E. One or more penetration flow paths with LPCI System or CS System testable check valve leakage limit not met.	E.1	Restore leakage rate to within limit.	72 hours
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.	12 hours
	AND		
	F.2	Be in MODE 4.	36 hours
G. Required Action and associated Completion Time of Condition A or B not met for PCIV(s) required to be OPERABLE during MODE 4 or 5.	G.1	Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	OR		
	G.2	Initiate action to restore valve(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.1	------------------NOTE------------------ Not required to be met when the 20 inch and 24 inch primary containment vent and 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 the full-flow line to Standby Gas Treatment (SGT) System is closed and one or more SGT System reactor building suction valves are open. -- Verify each 20 inch and 24 inch primary containment vent and 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 manual isolation 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 for MSIVs, is within limits.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV actuates to the isolation position on a simulated instrument line break.	In accordance with the Inservice Testing Program
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify the leakage rate of each air operated testable check valve associated with the LPCI and CS Systems vessel injection penetrations is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.3 Primary Containment Isolation Valves (PCIVs)

BASES

BACKGROUND

The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) to within limits. Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The OPERABILITY requirements for PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Therefore, the OPERABILITY requirements provide assurance that primary containment function assumed in the safety analyses will be maintained. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges and closed systems are considered passive devices. Check valves, and other automatic valves designed to close without operator action following an accident, are considered active devices. One or more barriers are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. When two or more barriers are provided, one of these barriers may be a closed system.

The reactor building-to-suppression chamber vacuum breakers serve a dual function, one of which is primary containment isolation. However, since the other safety function of the vacuum breakers would not be available if the normal PCIV actions were taken, the PCIV OPERABILITY requirements are not applicable to the reactor building-to-suppression chamber vacuum breakers valves. Similar surveillance requirements in the LCO for reactor building-to-suppression

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

are minimized. Of the events analyzed in Reference 1 for which the consequences are mitigated by PCIVs, the MSLB is the most limiting event due to radiological consequences to control room personnel. The closure time of the main steam isolation valves (MSIVs) is a significant variable from a radiological standpoint. The MSIVs are required to close within 3 to 5 seconds, after signal generation, since the 3 second closure time is assumed in the MSIV closure analysis (Ref. 2) and the 5 second closure time is consistent with or conservative to the times assumed in the MSLB analyses (Refs. 3 and 4). Likewise, it is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

The DBA analysis does not assume a specific closure time for primary containment isolation valves (PCIVs). The analysis assumes that the leakage from the primary containment is 1.5 percent primary containment air weight per day (L_a) at pressure P_a throughout the accident. The bases for PCIV closure times, and the specified valve closure times, are specified in UFSAR Section 7.3.3.1 and UFSAR Table 7.3-1 (Refs. 5 and 6), respectively.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the primary containment vent and purge valves. Two valves in series on each vent and purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred.

PCIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 7).

LCO

PCIVs form a part of the primary containment boundary. The PCIV safety function is related to minimizing the loss of reactor coolant inventory and establishing the primary containment boundary during a DBA.

The power operated, automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. The 20 and 24 inch vent and purge valves must be maintained closed or blocked to prevent full opening. While the reactor building-to-suppression chamber vacuum breakers isolate primary containment penetrations, they are excluded from this Specification. Controls on their isolation function are adequately addressed in LCO 3.6.1.6, "Reactor Building-to-Suppression Chamber Vacuum Breakers." The valves covered by this LCO are listed in Reference 8. The associated stroke time of each automatic PCIV is included in the Inservice Testing (IST) Program.

(continued)

BASES

LCO (continued)

The normally closed PCIVs are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves and devices are those listed in Reference 8.

MSIVs, Low Pressure Coolant Injection (LPCI) and Core Spray (CS) System air operated testable check valves must meet additional leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment," as Type B or C testing.

This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the primary containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, most PCIVs are not required to be OPERABLE and the primary containment vent and purge valves are not required to be normally closed in MODES 4 and 5. Certain valves, however, are required to be OPERABLE to prevent inadvertent reactor vessel draindown. These valves are those whose associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation." (This does not include the valves that isolate the associated instrumentation.)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow path(s) to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

A second Note has been added to provide clarification that, for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable,

(continued)

BASES

ACTIONS (continued)

since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable PCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable PCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are modified by Notes 3 and 4. Note 3 ensures that appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable PCIV (e.g., an Emergency Core Cooling System subsystem is inoperable due to a failed open test return valve). Note 4 ensures appropriate remedial actions are taken when the primary containment leakage limits are exceeded. Pursuant to LCO 3.0.6, these actions are not required even when the associated LCO is not met. Therefore, Notes 3 and 4 are added to require the proper actions be taken.

<u>A.1 and A.2</u>

With one or more penetration flow paths with one PCIV inoperable, except for inoperabilities due to MSIV leakage or LPCI or CS System air operated testable check valve leakage not within limit, the affected penetration flow paths must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available valve to the primary containment. The Required Action must be completed within the 4 hour Completion Time (8 hours for main steam lines). The Completion Time of 4 hours is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. For main steam lines, an 8 hour Completion Time is allowed. The Completion Time of 8 hours for the main steam lines allows a period of time to restore the MSIVs to OPERABLE status given the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration flow path(s) must be verified to be isolated on a periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following an accident, and no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those devices outside containment and capable of potentially being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside primary containment" is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. For the devices inside primary containment, the time period specified "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" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the devices and other administrative controls ensuring that device misalignment is an unlikely possibility.

Condition A is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two or more PCIVs. For penetration flow paths with one PCIV, Condition C provides the appropriate Required Actions. /T

Required Action A.2 is modified by two notes. Note 1 applies to isolation devices located in high radiation areas, and allows them to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to the isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing of components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

(continued)

BASES

ACTIONS (continued)

B.1

With one or more penetration flow paths with two or more PCIVs inoperable except for inoperabilities due to MSIV leakage or LPCI or CS System air operated testable check valve leakage not within limits, either the inoperable PCIVs must be restored to OPERABLE status or the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active component failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two or more PCIVs. For penetration flow paths with one PCIV, Condition C provides the appropriate Required Actions.

C.1 and C.2

With one or more penetration flow paths with one PCIV inoperable except for inoperabilities due to MSIV leakage or LPCI or CS System air operated testable check valve leakage not within limits, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active component failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. The Completion Time of 4 hours is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. The Completion Time of 72 hours for penetrations with a closed system is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. The closed system must meet the requirements of Reference 9. The Completion Time of 72 hours for EFCVs is also reasonable considering the instrument and the small pipe diameter of penetration (hence, reliability) to act as a penetration isolation boundary and the small pipe diameter of the affected penetrations. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration must be verified to be isolated on a

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following an accident are isolated. This Required Action does not require any testing or device manipulation. Rather, it involves verification, that those devices outside containment and capable of potentially being mispositioned are in the correct position. The Completion Time of once per 31 days for verifying each affected penetration is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to penetration flow paths with only one PCIV. For penetration flow paths with two or more PCIVs, Conditions A and B provide the appropriate Required Actions.

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

D.1

With any MSIV leakage rate not within limit, the assumptions of the safety analysis may not be met. Therefore, the leakage must be restored to within limit within 8 hours. Restoration can be accomplished by isolating the penetration that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway

(continued)

BASES

ACTIONS

D.1 (continued)

leakage of the two devices. The 8 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration, the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown, and the relative importance of MSIV leakage to the overall containment function.

E.1

With the one or more penetration flow paths with LPCI System or CS System air operated testable check valve leakage rate not within limits, the assumptions of the safety analysis may not be met. Therefore, the leakage must be restored to within limit within 72 hours. Restoration can be accomplished by isolating the penetration that caused the limit to be exceeded by use of one closed and de-activated automatic valve, or closed manual valve. When a penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 72 hour Completion Time is reasonable considering the time required to restore the leakage and the importance to maintain these penetrations available to perform the required function during a design basis accident.

F.1 and F.2

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

G.1 and G.2

If any Required Action and associated Completion Time cannot be met for PCIV(s) required to OPERABLE during MODE 4 or 5, the plant must be placed in a condition in which the LCO does not apply. Action must be immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended and valve(s) are restored to OPERABLE status. If suspending an OPDRV would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to immediately initiate action to restore the valve(s) to OPERABLE status. This allows RHR shutdown cooling to remain in service while actions are being taken to restore the valve.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1

This SR ensures that the primary containment vent and purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. The SR is modified by a Note stating that the SR is not required to be met when the vent and purge valves are open for the stated reasons. The Note states that these valves may be opened for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open, provided the full-flow 12 inch line (with valve 27MOV-120) to the SGT System is closed and one or more SGT System reactor building suction valves are open. This will ensure there is no damage to the filters if a LOCA were to occur with the vent and purge valves open since excessive differential pressure is not expected with the full-flow 12 inch line closed and one or more SGT System reactor building suction valves open. The 20 and 24 inch vent and purge valves are capable of closing against the dynamic effects of a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other PCIV requirements discussed in SR 3.6.1.3.2.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.2

This SR ensures that 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. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the primary containment boundary is within design limits.

This SR does not require any testing or valve manipulation. Rather, it involves verification that those isolation devices outside primary containment, and capable of being mispositioned, are in the correct position. Since verification of valve position for isolation devices outside primary containment is relatively easy, the 31 day Frequency was chosen to provide added assurance that the isolation devices are in the correct positions.

Two Notes have been added to this SR. The first Note allows valves and blind flanges located in high radiation areas to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable since the primary containment is inerted and access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these isolation devices, once they have been verified to be in the proper position, is low. A second Note has been included to clarify that PCIVs that are open under administrative controls are not required to meet the SR during the time that the PCIVs are open. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

SR 3.6.1.3.3

This SR ensures that each primary containment manual isolation valve and blind flange that is located inside primary containment and not locked, sealed or otherwise

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.3 (continued)

secured and is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the primary containment boundary is within design limits. For isolation devices inside primary containment, the Frequency defined as "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" is appropriate since these isolation devices are operated under administrative controls and the probability of their misalignment is low. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

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

SR 3.6.1.3.4

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.4 (continued)

operating experience that has demonstrated the reliability of the explosive charge continuity.

SR 3.6.1.3.5

Verifying the isolation time of each power operated, automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.6. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program. 1 J

SR 3.6.1.3.6

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

SR 3.6.1.3.7

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.1, "Primary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience has shown that these

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) is OPERABLE by verifying that the valve actuates to the isolation position on a simulated instrument line break. This SR provides assurance that the instrumentation line EFCVs will perform so that secondary containment will not be overpressurized during the postulated instrument line break (Ref. 10). The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

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

SR 3.6.1.3.10

The analyses in Reference 11 are based on leakage that is more than the specified leakage rate. Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ 25 psig. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.11

Surveillance of each air operated testable check valve associated with the LPCI and CS System vessel injection penetrations provides assurance that the resulting radiation dose rate that would result if the reactor coolant were released to the reactor building at the specified limit will be small (Ref. 12). The acceptance criteria for each air operated testable check valve associated with the LPCI and CS Systems vessel injection penetrations is < 10 gpm when hydrostatically tested at ≥ 1035 psig or < 10 scfm when pneumatically tested at ≥ 45 psig, at ambient temperature (Ref. 12). The leakage rates must be demonstrated in accordance with the leakage rate test Frequency required by the Primary Containment Leakage Rate Testing Program.

REFERENCES

1. UFSAR, Section 14.6.
2. UFSAR, Section 14.5.2.3.
3. UFSAR, Section 6.5.3.2.
4. UFSAR, Section 14.8.2.1.2.
5. UFSAR, Section 7.3.3.1.
6. UFSAR, Table 7.3-1.
7. 10 CFR 50.36(c)(2)(ii).
8. Technical Requirements Manual.
9. UFSAR, Section 5.2.3.5.
10. UFSAR, Section 16.3.2.5.
11. UFSAR, Section 14.8.2.1.1.
12. NRC Letter to NYPA, November 9, 1978 NRC Safety Evaluation Supporting Amendment 40 to the Facility Operating License No. DPR-59.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Pressure

BASES

BACKGROUND	The drywell pressure is limited during normal operations to preserve the initial conditions assumed in the accident analysis for a Design Basis Accident (DBA) or loss of coolant accident (LOCA). DB1
APPLICABLE SAFETY ANALYSES	Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs (Ref. 1). Among the inputs to the DBA is the initial primary containment internal pressure (Ref. 1) 2 and 3. Analyses assume an initial drywell pressure of 0.75 1.95 psig. DB2 This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell internal pressure does not exceed the ~~maximum allowable~~ drywell design pressure of PA1 of 62 56 psig. DB2 The maximum calculated drywell pressure occurs during the reactor blowdown phase of the DBA, which assumes an instantaneous recirculation line break. The calculated peak drywell pressure for this limiting event is 57.5 41.2 DB2 psig (Ref. 1 4). DB1 Drywell pressure satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 5). X1
LCO	In the event of a DBA, with an initial drywell pressure ≤ 0.75 1.95 DB2 psig, the resultant peak drywell accident pressure will be maintained below the drywell ~~design~~ maximum allowable pressure. PA1
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell pressure within limits is not required in MODE 4 or 5.

(continued)

Revision J

X1 DB1 Insert REF

2. NEDO-24578, Revision 0, Mark I Containment Program Plant Unique Load Definition, James A. FitzPatrick Nuclear Power Plant, March 1979.

3. UFSAR, Section 16.9.3.5.

4. UFSAR, Section 16.9.3.5.1.3.

5. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Pressure

BASES

BACKGROUND

The drywell pressure is limited during normal operations to preserve the initial conditions assumed in the accident analysis for a Design Basis Accident (DBA) or loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs (Ref. 1). Among the inputs to the DBA is the initial primary containment internal pressure (Refs. 1, 2 and 3). Analyses assume an initial drywell pressure of 1.95 psig. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell internal pressure does not exceed the drywell design pressure of 56 psig.

The maximum calculated drywell pressure occurs during the reactor blowdown phase of the DBA, which assumes an instantaneous recirculation line break. The calculated peak drywell pressure for this limiting event is 41.2 psig (Ref. 4).

Drywell pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

LCO

In the event of a DBA, with an initial drywell pressure ≤ 1.95 psig, the resultant peak drywell accident pressure will be maintained below the maximum allowable drywell pressure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell pressure within limits is not required in MODE 4 or 5.

(continued)

BASES

REFERENCES (continued)

3. UFSAR, Section 16.9.3.5.

4. UFSAR, Section 16.9.3.5.1.3.

5. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Drywell Air Temperature

BASES

BACKGROUND

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 1) and 2. Analyses assume an initial average drywell air temperature of [135]°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature and pressure does not exceed the ~~maximum allowable temperature of [340]°F~~ design pressure of 56 psig coincident with a design temperature of 309°F (Ref. 2 3). Exceeding these design ~~temperature~~ limitations may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the ~~accident~~ spectrum of break sizes.

DB1 DB2 5 drywell DB1 DB1 5 PA1

Drywell air temperature satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 4).

X1 X1 5

LCO

In the event of a DBA, with an initial drywell average air temperature and pressure less than or equal to the LCO temperature limit are, the resultant peak accident temperature is maintained ~~below~~ within the drywell design ~~temperature~~ limits and within the environmental qualification envelope of the equipment in the drywell. As a result, the ability of primary containment to perform its design function is ensured.

DB1 DB1

(continued)

Revision 5

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell average air temperature within the limit is not required in MODE 4 or 5.

ACTIONS

A.1

With drywell average air temperature not within the limit of the LCO, drywell average air temperature must be restored within 8 hours. The Required Action is necessary to return operation to within the bounds of the primary containment analysis. The 8 hour Completion Time is acceptable, considering the sensitivity of the analysis to variations in this parameter, and provides sufficient time to correct minor problems.

PA2

B.1 and B.2

the

If the drywell average air temperature cannot be restored to within limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1

PA3

Verifying that the drywell average air temperature is within the LCO limit ensures that operation remains within the limits assumed for the primary containment analyses. Drywell air temperature is monitored in ~~all quadrants~~ various areas and at various elevations (referenced to mean sea level). Due to the shape of the drywell, a volumetric average is used to determine an accurate representation of the actual average temperature.

(continued)

Revision J

BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.5.1 (continued)

The 24 hour Frequency of the SR was developed based on operating experience related to drywell average air temperature variations and temperature instrument drift during the applicable MODES and the low probability of a DBA occurring between surveillances. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal drywell air temperature condition.

PA3 14.6.1.3.3 DB3

REFERENCES

1. U FSAR, Section ~~6.2~~.

3 ~~2~~. U FSAR, Section ~~6.2.1.4.1~~. 16.7.3.2.3

DB1

~~3. FSAR, Section [6.2.1.4.5].~~

PA4

4. 10 CFR 50.36 (c)(2)(ii). 1 J

X1

1 J

Insert REF

DB1

Revision J

DB1

Insert REF

2. GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperatures Analysis, August 1996.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Drywell Air Temperature

BASES

BACKGROUND

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Refs. 1 and 2). Analyses assume an initial average drywell air temperature of 135°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature and pressure do not exceed the drywell design pressure of 56 psig coincident with a design temperature of 309°F (Ref. 3). Exceeding these design limitations may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the spectrum of break sizes.

Drywell air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

In the event of a DBA, with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature and pressure are maintained within the drywell design limits and within the environmental qualification envelope of the equipment in the drywell. As a result, the ability of primary containment to perform its design function is ensured.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell average air temperature within the limit is not required in MODE 4 or 5.

ACTIONS

A.1

With drywell average air temperature not within the limit of the LCO, drywell average air temperature must be restored within 8 hours. The Required Action is necessary to return operation to within the bounds of the primary containment analysis. The 8 hour Completion Time is acceptable, considering the sensitivity of the analysis to variations in this parameter, and provides sufficient time to correct minor problems.

B.1 and B.2

If the drywell average air temperature cannot be restored to within the limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1

Verifying that the drywell average air temperature is within the LCO limit ensures that operation remains within the limits assumed for the primary containment analyses. Drywell air temperature is monitored in various areas and at various elevations (referenced to mean sea level). Due to the shape of the drywell, a volumetric average is used to determine an accurate representation of the actual average temperature. |J

(continued)

BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.5.1 (continued)

The 24 hour Frequency of the SR was developed based on operating experience related to drywell average air temperature variations and temperature instrument drift during the applicable MODES and the low probability of a DBA occurring between surveillances. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal drywell air temperature condition.

REFERENCES

1. UFSAR, Section 14.6.1.3.3.
2. GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, August 1996.
3. UFSAR, 16.7.3.2.3.
4. 10 CFR 50.36(c)(2)(ii).

DISCUSSION OF CHANGES
ITS: 3.6.1.6 - REACTOR BUILDING-TO-SUPPRESSION CHAMBER
VACUUM BREAKERS

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted that do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.7.A.4.b allows 7 days to restore an inoperable reactor building-to-suppression chamber vacuum breaker provided primary containment integrity is maintained. ITS 3.6.1.6, ACTIONS A and C, stipulate restoration within 72 hours of the affected vacuum breaker valves in the reactor building-to-suppression chamber line(s) provided at least one valve in each line is closed and as long as one line is Operable for the opening function, respectively. This represents an additional restriction on plant operation and constitutes a more restrictive change necessary to ensure timely action is taken to restore the capability to withstand a single failure in the reactor building-to-suppression chamber vacuum breaker relief system.

M2 SR 3.6.1.6.1 is proposed to be added to CTS 4.7.A.4 to verify that the reactor building-to-suppression chamber vacuum breakers are closed. This SR serves to provide verification that a potential breach in the primary containment boundary is not present. The addition of new Surveillance Requirements constitutes a more restrictive change but intended to ensure safe operation.

M3 SR 3.6.1.6.4 is proposed to be added to CTS 4.7.A.4 to verify that the reactor building-to-suppression chamber self actuating vacuum breakers (27VB-6 and 27VB-7) are capable of opening at a differential pressure of $\leq$ 0.5 psid which will ensure the safety analysis assumptions are met. Since there is no explicit requirement for the self actuating vacuum breakers, this change is considered more restrictive but safer on plant operations since it will convey the proper functioning status of each vacuum breaker. J

SURVEILLANCE REQUIREMENTS (continued)

6 PA1 SURVEILLANCE		FREQUENCY
SR 3.6.1.7.8 4	Verify the opening setpoint of each self actuating vacuum breaker is ≤ [0.5] psid.	[18] 24 months X1

CLB 2 CLB 2 DB 2

[M3]

J

SR 3.6.1.6.3	perform a CHANNEL CALIBRATION of each air operated vacuum breaker differential pressure instrument channel and verify the setpoint is ≤ 0.5 psid.	92 days

4.7.A.4.b
3.7.A.4.b
[M5]

CLB 2

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.6.1.6 - REACTOR BUILDING-TO-SUPPRESSION CHAMBER
VACUUM BREAKERS

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The brackets have been removed and the Frequency of ISTS 3.6.1.7.2 (ITS 3.6.1.6.2) has been changed to "In accordance with the Inservice Testing Program," consistent with the current licensing basis in CTS 4.7.A.4.a.

CLB2 ITS SR 3.6.1.6.3 has been added to help ensure the OPERABILITY of the differential pressure instrumentation channels. This requirement is consistent with CTS 3.7.A.4.a and 4.7.A.4.b. Subsequent Surveillances have been renumbered as necessary. In addition, ISTS SR 3.6.1.7.3 (ITS SR 3.6.1.6.4) has been modified so that it will only be applicable to the self actuating vacuum breakers since ITS SR 3.6.1.6.1, SR 3.6.1.6.2 and SR 3.6.1.6.3 will ensure the air-operated vacuum breakers function properly.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 JAFNPP will not adopt ISTS 3.6.1.6. As a result, ISTS 3.6.1.7 has been renumbered.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the words "or more" deleted since the plant specific design only includes two lines.

DB2 The brackets have been removed and the proper plant specific value has been provided.

J

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DBI unless otherwise noted

PA1

BASES

BACKGROUND (continued)

Low spray temperatures and atmospheric conditions that yield the minimum amount of contained noncondensible gases are assumed for conservatism.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the reactor building-to-suppression chamber vacuum breakers are presented in Reference 1 as part of the accident response of the containment systems. Internal (suppression chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls, which form part of the primary containment boundary.

PA3

The safety analyses assume the ~~external~~ reactor building-to-suppression chamber vacuum breakers to be closed initially ~~and to be fully open at [0.5] psid~~ (Ref. 1). Additionally, ~~of the two~~ reactor building-to-suppression chamber vacuum breakers in each line, ~~one is~~ one or both are assumed to fail in a closed position, ~~to satisfy~~ the single active failure criterion is met. ~~Design Basis Accident (DBA) analyses require the vacuum breakers to be closed initially and to remain closed and leak tight with positive primary containment pressure.~~ Therefore,

1 J

~~Five~~ Several cases were considered in the safety analyses to determine the ~~adequacy of the external vacuum breakers~~ maximum negative pressure differential between the containment and reactor building assuming the reactor building-to-suppression chamber vacuum breakers remain closed (Ref. 1):

a. A small break loss of coolant accident followed by actuation of ~~both primary~~ one Residual Heat Removal (RHR) containment spray loop~~s~~;

b. Inadvertent actuation of one ~~primary~~ RHR containment spray loop during normal operation;

~~c. Inadvertent actuation of both primary containment spray loops during normal operation;~~

~~d. A postulated DBA assuming Emergency Core Cooling Systems (ECCS) runout flow with a condensation effectiveness of 50%; and~~

~~e.~~ c. A ~~postulated DBA assuming ECCS runout flow with a condensation effectiveness of 100%~~ large break loss of coolant accident followed by actuation of one RHR containment spray loop.

The results of these ~~five~~ cases show that the ~~external~~ reactor building-to-suppression chamber vacuum breakers~~, with an opening setpoint of [0.5] psid, are~~ are not required to mitigate the consequences of any DBA since the maximum resulting negative differential pressure is 1.92 psid (case a) which is below the design limit of 2 psid.

(continued)

Revision J

DB1

Insert ASA

However, to ensure the resulting negative pressure is minimized, the reactor building-to-suppression chamber vacuum breakers are included in the design and set to ensure the valves start to open at ≤ 0.5 psid. J

BASES 6 PA1

SURVEILLANCE REQUIREMENTS

SR 3.6.1.~~7~~.1 (continued)

judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

Two Notes are added to this SR. The first Note allows reactor building-to-suppression chamber vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers. The second Note is included to clarify that vacuum breakers open due to an actual differential pressure are not considered as failing this SR. PA4 J

SR 3.6.1.~~7~~ 6.2 PA1

Each vacuum breaker must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This ensures that the safety analysis assumptions are valid. The [92 day] Frequency of this SR ~~was developed based upon~~ is in accordance with the Inservice Testing Program ~~requirements to perform valve testing at least once every [92] days~~. PA3 CLB1

INSERT SR 3.6.1.6.3 CLB2

SR 3.6.1.~~7~~ 6.~~3~~ 4 PA1

Demonstration of each self-actuating vacuum breaker design function opening setpoint is necessary to ensure that the ~~safety analysis~~ assumption regarding vacuum breaker ~~full~~ opening differential pressure of ≤ [0.5] psid is valid. ~~The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~ For this unit, the ~~[18]~~ 24 month Frequency ~~has been shown to be acceptable, based on operating experience, and~~ is further justified because ~~other surveillances~~ SR 3.6.1.6.2 is performed at shorter Frequencies that convey the proper functioning status of each self-actuating vacuum breaker. CLB2 DB1 DB2 PA4 J X2 DB5 CLB2 DB5

Insert SR 3.6.1.6.4

REFERENCES

~~1. FSAR, Section [6.2].~~ Insert Ref. DB6

Revision J

Insert SR 3.6.1.6.3 CLB2

SR 3.6.1.6.3

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

The Frequency of SR 3.6.1.6.3 is based on the assumptiom of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

Insert SR 3.6.1.6.4 DB5

While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency, which is based on the operating cycle.

Insert Ref DB6

1. Design Basis Document-016A, Section 5.2.10, Maximum Design Negative Pressure for Containment.

2. 10 CFR 50.36 (c)(2)(ii). - X1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.6.3	Perform a CHANNEL CALIBRATION of each air operated vacuum breaker differential pressure instrument channel and verify the setpoint is ≤ 0.5 psid.	92 days
SR 3.6.1.6.4	Verify the opening setpoint of each self actuating vacuum breaker is ≤ 0.5 psid.	24 months

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.6 Reactor Building-to-Suppression Chamber Vacuum Breakers

BASES

BACKGROUND

The function of the reactor building-to-suppression chamber vacuum breakers is to relieve vacuum when primary containment depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building-to-suppression chamber vacuum breakers and through the suppression-chamber-to-drywell vacuum breakers. The design of the reactor building-to-suppression chamber vacuum relief system consists of four vacuum breakers (two parallel sets of 100% capacity vacuum breaker pairs, each set consisting of a self-actuating vacuum breaker and an air operated vacuum breaker), located in two lines. The air operated vacuum breakers are actuated by differential pressure switches and can be remotely operated from the relay room. The self-actuating vacuum breakers function similar to a check valve. The two vacuum breakers in series must be closed to maintain a leak tight primary containment boundary.

A negative differential pressure across the drywell wall is caused by rapid depressurization of the drywell. Events that cause this rapid depressurization are cooling cycles, inadvertent primary containment spray actuation, and steam condensation in the event of a primary system rupture. Reactor building-to-suppression chamber vacuum breakers prevent an excessive negative differential pressure across the primary containment boundary. Cooling cycles result in minor pressure transients in the drywell, which occur slowly and are normally controlled by ventilation equipment. J
Inadvertent spray actuation results in a more significant negative pressure transient.

The reactor building-to-suppression chamber vacuum breakers are sized to mitigate any depressurization transient and limit the maximum negative containment (drywell and suppression chamber) pressure to within design limits. The maximum depressurization rate is a function of the primary containment spray flow rate and temperature and the assumed initial conditions of the primary containment atmosphere.

(continued)

BASES

BACKGROUND (continued)	Low spray temperatures and atmospheric conditions that yield the minimum amount of contained noncondensible gases are assumed for conservatism.

APPLICABLE SAFETY ANALYSES

Suppression chamber-to-drywell and reactor building-to-suppression chamber vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls, which form part of the primary containment boundary.

The safety analyses assume the reactor building-to-suppression chamber vacuum breakers to be closed initially (Ref. 1). Additionally, one or both reactor building-to-suppression chamber vacuum breakers in each line are assumed to fail in a closed position. Therefore, the single active failure criterion is met.

Several cases were considered in the safety analyses to determine the maximum negative pressure differential between the containment and reactor building assuming the reactor building-to-suppression chamber vacuum breakers remain closed (Ref. 1):

a. A small break loss of coolant accident followed by actuation of one Residual Heat Removal (RHR) containment spray loop;

b. Inadvertent actuation of one RHR containment spray loop during normal operation;

c. A large break loss of coolant accident followed by actuation of one RHR containment spray loop.

The results of these cases show that the reactor building-to-suppression chamber vacuum breakers are not required to mitigate the consequences of any DBA since the maximum resulting negative differential pressure is 1.92 psid (case a) which is below the design differential pressure limit of 2 psid. However, to ensure the resulting negative pressure is minimized, the reactor building-to-suppression chamber vacuum breakers are included in the design and set to ensure the valves start to open at ≤ 0.5 psid. | J

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The reactor building-to-suppression chamber vacuum breakers satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

LCO

All reactor building-to-suppression chamber vacuum breakers are required to be OPERABLE to ensure the primary containment design differential pressure limit is not challenged. This requirement ensures both vacuum breakers in each line (self-actuated vacuum breaker and air operated vacuum breaker) will open to relieve a negative pressure in the suppression chamber. This LCO also ensures that the two vacuum breakers in each of the two lines from the reactor building to the suppression chamber airspace are closed (except during testing or when performing their intended function).

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture, which purges the drywell of air and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell, which after the suppression chamber-to-drywell vacuum breakers open (due to differential pressure between the suppression chamber and drywell) would result in depressurization of the suppression chamber. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3. Excessive negative pressure inside primary containment could also occur due to inadvertent initiation of the RHR Containment Spray System. J

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining reactor building-to-suppression chamber vacuum breakers OPERABLE is not required in MODE 4 or 5.

(continued)

BASES

ACTIONS (continued)

D.1

With two lines with one or more vacuum breakers inoperable for opening, the primary containment boundary is intact. However, in the event of a containment depressurization, the vacuum relief function of the vacuum breakers is lost. Therefore, all vacuum breakers in one line must be restored to OPERABLE status within 1 hour. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, which requires that primary containment be restored to OPERABLE status within 1 hour.

E.1 and E.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

Each vacuum breaker is verified to be closed to ensure that a potential breach in the primary containment boundary is not present. This Surveillance may be performed by observing local or remote indications of vacuum breaker position. Position indications of the air operated vacuum breakers are available in the control and relay rooms while position indications of the self actuating vacuum breakers are only available in the relay room. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

Two Notes are added to this SR. The first Note allows reactor building-to-suppression chamber vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers. The second Note is included to clarify that vacuum breakers open due to an actual differential pressure are not considered as failing this SR.

SR 3.6.1.6.2

Each vacuum breaker must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This ensures that the safety analysis assumptions are valid. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.1.6.3

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

The Frequency of SR 3.6.1.6.3 is based on the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.6.1.6.4

Demonstration of each self-actuating vacuum breaker opening setpoint is necessary to ensure that the design function regarding vacuum breaker opening differential pressure of $\leq$ 0.5 psid is valid. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency, which is based on the operating cycle. The 24 month Frequency is further justified because SR 3.6.1.6.2 is performed at a shorter Frequency that conveys the proper functioning status of each self-actuating vacuum breaker.

(continued)

3.6 CONTAINMENT SYSTEMS

[3.7.A.5] 3.6.1.8 Suppression Chamber-to-Drywell Vacuum Breakers 7 PA1

[3.7.A.5.a] LCO 3.6.1.8 Each [Nine] suppression chamber-to-drywell vacuum breakers shall be OPERABLE for opening. 1 J

DB1

~~AND~~

~~[Twelve] suppression chamber-to-drywell vacuum breakers shall be closed, except when performing their intended function.~~

[3.7.A.5.a] APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.7.A.5.c] [M2] [3.7.A.5.g] A. One ~~required~~ (DB2) suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1 Restore ~~one~~ the (DB2) vacuum breaker to OPERABLE status.	72 hours
[3.7.A.5.g] [M3] B. One suppression chamber-to-drywell vacuum breaker not closed.	B.1 Close the open vacuum breaker.	2 hours
[3.7.A.8] [M4] [L2] C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

Revision J

PA1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
[4.7.A.5.a] [L5] SR 3.6.1.~~8~~ 7.2 (PA1) Perform a functional test of each ~~required~~ (DB2) vacuum breaker.	~~31 days~~ (X3) In accordance with the Inservice Testing Program ~~AND~~ ~~Within 12 hours after any discharge of steam to the suppression chamber from the S/RVs~~ ~~AND~~ ~~Within 12 hours following an operation that causes any of the vacuum breakers to open~~ (CLB2)
[3.7.A.5.f] [4.7.A.5.g] SR 3.6.1.~~8~~ 7.3 (PA1) Verify the opening setpoint of each ~~required~~ (DB2) vacuum breaker is ≤ [0.5] psid. (DB3)	~~18~~ 24 months (CLB3)

REVISION ~~E~~ J

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The JAFNPP design includes 5 suppression chamber-to-drywell vacuum breakers. The accident analysis requires 4 vacuum breakers to function. Therefore, to satisfy the single failure criteria all vacuum breakers must be Operable to satisfy the LCO. The ISTS LCO 3.6.1.8 has been reworded as required (ITS LCO 3.6.1.7). The ISTS LCO 3.6.1.8 detail that the valve must be closed except when performing their intended function has been moved to ISTS SR 3.6.1.8.1 as a Note (ITS SR 3.6.1.7.1 Note 2). Therefore, the first Note has been renumbered as required. This format change is consistent with the format of ISTS 3.6.1.7 for reactor building-to-suppression chamber vacuum breakers.

DB2 The term "required" in ITS 3.6.1.7 Condition A, SR 3.6.1.7.2, and SR 3.6.1.7.3 is not needed since all vacuum breakers must be OPERABLE and closed.

DB3 The brackets have been removed and the proper plant specific value has been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 Not used.

X2 The second Frequency of ITS SR 3.6.1.7.1 (NUREG SR 3.6.1.8.1) is being deleted. The Suppression chamber-to-drywell vacuum breakers have position indication for each valve in the relay room and when one or more of the valves is not fully closed Control Room Annunciator 09-3-3-39 is actuated to alert the Control Room operators of the condition. In addition, drywell-to-suppression chamber differential pressure is maintained in accordance with ITS 3.6.2.4 during most of the time period that the vacuum breakers are required to be OPERABLE (and normally closed). Maintenance of the differential pressure results in a closing force of more than 1000 pounds on each valve disk to keep them closed. Further, the valve seat is at an angle of approximately 25 degrees from

BASES

BACKGROUND (continued)

less than the suppression chamber pressure, there will be an increase in the vent waterleg. This will result in an increase in the water clearing inertia in the event of a postulated LOCA, resulting in an increase in the peak drywell pressure. This in turn will result in an increase in the pool swell dynamic loads. The ~~internal~~ suppression chamber-to drywell vacuum breakers limit the height of the waterleg in the vent system during ~~normal operation~~. DB1

time periods when drywell-to suppression chamber differential pressure is not required or is not maintained within the limits specified in LCO 3.6.2.4, "Drywell-to-Suppression Chamber Differential Pressure" CLB5

APPLICABLE SAFETY ANALYSES

PA3 PA3

Analytical methods and assumptions involving the suppression chamber-to-drywell vacuum breakers are ~~presented in Reference 1~~ analyses USAR PA3 as part of the accident ~~response~~ of the primary containment systems. ~~Internal (suppression chamber-to-drywell)~~ and ~~external (~~reactor building-to-suppression chamber~~)~~ vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls that form part of the primary containment boundary. DB1 DB1

The safety analyses assume that the ~~internal~~ suppression chamber-to-drywell vacuum breakers are closed initially and ~~are fully open at a differential pressure of [0.5] psid (Ref. 1)~~ start to 1 DB3. Additionally, ~~3~~ 5 of the ~~12~~ ~~internal~~ vacuum breakers ~~are~~ is assumed to fail in a closed position (Ref. 1). DB3 and 2 The results of the analyses show that the design differential pressure is not exceeded even under the worst case accident scenario. PA3 The vacuum breaker opening differential pressure setpoint and the requirement that ~~[9] of [12]~~ all vacuum breakers be OPERABLE ~~are a result of the requirement placed on the vacuum breakers to limit the vent system waterleg height. The total cross sectional area of the main vent system between the drywell and suppression chamber needed to fulfill this requirement has been established as a minimum of [51.5] times the total break area (Ref. 1). In turn, the vacuum relief capacity between the drywell and suppression chamber should be [1/16] of the total main vent cross sectional area, with the valves set to operate at [0.5] psid differential pressure.~~ DB2 PA3 (the additional vacuum breaker is required to meet the single failure criterion) PA3

Insert ASA DB3

Design Basis Accident (DBA) analyses ~~require~~ assume the vacuum breakers to be closed initially and to remain closed and leak tight, ~~with~~ until the suppression pool is at a positive pressure relative to the drywell. DB3

(continued)

Revision J

DB3

Insert ASA

The cross sectional area of the vacuum breakers are sized on the basis of the Bodega Bay pressure suppression system tests. The vacuum breaker capacity selected on this test basis is more than adequate to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to a value which is within the suppression system design values (Refs. 3 and 4). 5

BASES

APPLICABLE SAFETY ANALYSES (continued)

The suppression chamber-to-drywell vacuum breakers satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii) (Ref. 5) X1

LCO

A11 DB2

~~Only [9] of the [12]~~ All vacuum breakers must be OPERABLE for opening. All suppression chamber-to-drywell vacuum breakers~~, however,~~ are also required to be closed (except during testing or when the vacuum breakers are performing their intended design function). The vacuum breaker OPERABILITY requirement provides assurance that the drywell-to-suppression chamber negative differential pressure remains below the design value. The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur.

APPLICABILITY

DB4 PA3

In MODES 1, ~~2,~~ and 3, the ~~Suppression Pool Spray System~~ is required to be OPERABLE to mitigate the effects of a DBA. DB4 PA3

Excessive negative pressure inside the drywell could also occur due to inadvertent actuation of ~~this system~~. The vacuum breakers, ~~therefore~~, are required to be OPERABLE in MODES 1, 2, and 3, when the Suppression Pool Spray System is required to be OPERABLE, to mitigate the effects of inadvertent actuation of the Suppression Pool Spray System. DB4

~~Also,~~ In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall, caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture that purges the drywell of air and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES; therefore, maintaining suppression chamber-to-drywell vacuum breakers OPERABLE is not required in MODE 4 or 5.

DB4 PA3 the RHR Containment Spray System during normal operation

(continued)

REVISION ~~E~~ I

BASES (continued)

ACTIONS

A.1

DB7 negative drywell-to-suppression chamber PA3 DB2 four PA3 active DB2 five DB7

With one of the ~~required~~ vacuum breakers inoperable for opening (e.g., the vacuum breaker is not open and may be stuck closed or not within its opening setpoint limit, so that it would not function as designed during an event that depressurized the drywell), the remaining ~~[eight]~~ OPERABLE vacuum breakers are capable of providing the vacuum relief function. However, overall system reliability is reduced because a single failure in one of the remaining vacuum breakers could result in an excessive ~~suppression chamber-to-drywell~~ differential pressure during a DBA. Therefore, with one of the ~~[nine]~~ ~~required~~ vacuum breakers inoperable, 72 hours is allowed to restore ~~at least one of~~ the inoperable vacuum breaker~~s~~ to OPERABLE status so that plant conditions are consistent with those assumed for the design basis analysis. The 72 hour Completion Time is considered acceptable due to the low probability of an event ~~in which~~ the remaining vacuum breaker capability ~~would not be adequate~~.

PA3 occurring that would require X3 PA3

B.1

primary containment DB6 PA3

An open vacuum breaker allows communication between the drywell and suppression chamber airspace, and, as a result, there is the potential for ~~suppression chamber~~ overpressurization due to this bypass leakage if a LOCA were to occur. Therefore, the open vacuum breaker must be closed. A short time is allowed to close the vacuum breaker due to the low probability of an event that would pressurize primary containment. If vacuum breaker position indication is not reliable, an alternate method of verifying that the vacuum breakers are closed ~~is to verify that a differential pressure of [0.5] psid between the suppression chamber and drywell is maintained for 1 hour without makeup~~. The required 2 hour Completion Time is considered adequate to perform this test.

DB6

is to verify the bypass leakage between the drywell and suppression chamber is within the limits of SR 3.6.1.1.2 or by local observation.

C.1 and C.2

any Required Action and associated Completion Time cannot be met, PA4

If ~~the inoperable suppression chamber-to-drywell vacuum breaker cannot be closed or restored to OPERABLE status within the required Completion Time~~, the plant must be brought to a MODE in which the LCO does not apply. To

X2

(continued)

If the leak test fails, not only must this ACTION be taken (close the open vacuum breaker within the required Completion Time), but also the appropriate Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," must be entered.

Revision J

BASES

ACTIONS

C.1 and C.2 (continued)

achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.1 7 PA1

Each vacuum breaker is verified closed to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker position indication or by ~~verifying that a differential pressure of [0.5] psid between the suppression chamber and drywell is maintained for 1 hour without makeup~~. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience. ~~This verification is also required within 2 hours after any discharge of steam to the suppression chamber from the safety/relief valves or any operation that causes the drywell-to-suppression chamber differential pressure to be reduced by ≥ [0.5] psid.~~

DB3 local or relay room

DB6 performing SR 3.6.1.1.2 the bypass leakage test.

X4 Insert SR 3.6.1.7.1

X5

If the bypass test fails, not only must the vacuum breaker(s) be considered open and the appropriate Conditions and Required Actions of this LCO be entered, but also the appropriate Conditions and Required Actions of LCO 3.6.1.1 must be entered.

X2

~~A~~ Two ~~Note~~ Notes ~~is~~ are added to this SR. The first Note ~~which~~ allows suppression chamber-to-drywell vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers.

DB2 J DB2

The second Note is included to clarify that vacuum breakers open due to an actual differential pressure are not considered as failing this SR.

J

SR 3.6.1.8.2 7 PA1

Each ~~required~~ vacuum breaker must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This ensures that the safety analysis assumptions are valid. The 31 day Frequency of this SR ~~was developed based on~~ is in accordance with the Inservice Testing Program ~~requirements to perform valve testing at least once every 92 days. A 31 day Frequency was chosen to provide~~

DB7 X6

(continued)

Revision J

BASES

7 PA1

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.2 (continued)

additional assurance that the vacuum breakers are OPERABLE, since they are located in a harsh environment (the suppression chamber airspace). In addition, this functional test is required within 12 hours after either a discharge of steam to the suppression chamber from the safety/relief valves or after an operation that causes any of the vacuum breakers to open.

X6 CLB4 CLB2

7 PA1

SR 3.6.1.8.3

Verification of the vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption ing regarding vacuum breaker full open differential pressure of [0.5] psid is valid. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. For this facility, the [18] month Frequency has been shown to be acceptable, based on operating experience, and is further justified because of other surveillances performed at shorter Frequencies that convey the proper functioning status of each vacuum breaker.

DB3 1 J DB8 CLB3 PA3 24 4 a PA3 SR 3.6.1.7.2 is 5

REFERENCES

PA5 1. U FSAR, Section [6.2]. 14.6.1.3.3 DB5

2. UFSAR, Section 5.2.3.6.
3. UFSAR, Section 5.2.4.2.
4. Preliminary Hazards Summary Report, Bodega Bay Atomic Park Unit Number 1, Docket No. 50-205, Appendix I, December 28, 1962.
5. 10 CFR 50.36 (c)(2)(ii).

DB3 1 J 1 J 1 J 1 J X1

Revision I

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB6 The appropriate plant specific alternative methods for verification that the vacuum breakers are closed has been included in ITS 3.6.1.7 ACTION B and SR 3.6.1.7.1.

DB7 The term "required" and "at least one of" in ITS 3.6.1.7 Condition A and the term "required" in ITS SR 3.6.1.7.2 is not needed since all vacuum breakers must be OPERABLE and closed. 5

DB8 The brackets have been removed and the proper plant specific value has been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

X2 This test ensures the suppression chamber-to-drywell vacuum breakers are closed. The suppression chamber-to-drywell vacuum breaker instrumentation may be inoperable or undergoing maintenance and therefore proper suppression chamber-to-drywell vacuum breaker position indication may not be available at the time of the performance of SR 3.6.1.7.1. If excessive leakage existed, the suppression chamber and drywell pressure instrumentation would have indicated whether the primary containment was inoperable. ITS SR 3.0.1 will require all SRs to 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. Therefore, as a result of ITS SR 3.6.1.1.2, the associated ACTIONS of ITS 3.6.1.1 (1 hours for primary containment inoperability), and SR 3.0.1, the 12 hour allowance is acceptable since entry into ITS 3.6.1.1 ACTION A will be

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. 1/5

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1 Restore the vacuum breaker to OPERABLE status.	72 hours
B. One suppression chamber-to-drywell vacuum breaker not closed.	B.1 Close the open vacuum breaker.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 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.	In accordance with the Inservice Testing Program
SR 3.6.1.7.3	Verify the opening setpoint of each vacuum breaker is ≤ 0.5 psid.	24 months

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.7 Suppression Chamber-to-Drywell Vacuum Breakers

BASES

BACKGROUND

The function of the suppression chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell. There are 5 external vacuum breakers located on the external lines connecting the top of the suppression chamber with drywell vent pipes, which allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Therefore, suppression chamber-to-drywell vacuum breakers prevent an excessive negative differential pressure across the wetwell drywell boundary. Each vacuum breaker is a self-actuating valve, similar to a check valve, which can be manually operated locally for testing purposes.

A negative differential pressure across the drywell wall is caused by rapid depressurization of the drywell. Events that cause this rapid depressurization are cooling cycles, drywell spray actuation, and steam condensation from sprays or subcooled reflood water in the event of a primary system rupture. Cooling cycles result in minor pressure transients in the drywell that occur slowly and are normally controlled by ventilation equipment. Spray actuation or the spilling of subcooled water out of a break results in more significant pressure transients and becomes important in sizing the suppression chamber-to-drywell vacuum breakers.

In the event of a primary system rupture, steam condensation within the drywell results in the most severe pressure transient. Following a primary system rupture, the gas mixture in the drywell is purged into the suppression chamber free airspace, leaving the drywell full of steam. Subsequent condensation of the steam can be caused in two possible ways, namely, Emergency Core Cooling Systems flow out of a line break, or Residual Heat Removal (RHR) Containment Spray System actuation following a loss of coolant accident (LOCA). These two cases determine the maximum depressurization rate of the drywell.

In addition, the waterleg in the Mark I Vent System downcomers are controlled by the drywell-to-suppression chamber differential pressure. If the drywell pressure is

(continued)

BASES

BACKGROUND (continued)

less than the suppression chamber pressure, there will be an increase in the vent waterleg. This will result in an increase in the water clearing inertia in the event of a postulated LOCA, resulting in an increase in the peak drywell pressure. This in turn will result in an increase in the pool swell dynamic loads. The suppression chamber-to-drywell vacuum breakers may limit the height of the waterleg in the vent system during time periods when drywell-to-suppression chamber differential pressure is not required or is not maintained within the limits specified in LCO 3.6.2.4, "Drywell-to-Suppression Chamber Differential Pressure."

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the suppression chamber-to-drywell vacuum breakers are used as part of the accident analyses of the primary containment systems. Suppression chamber-to-drywell and reactor building-to-suppression chamber vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls that form part of the primary containment boundary.

The safety analyses assume that the suppression chamber-to-drywell vacuum breakers are closed initially and start to open at a differential pressure of 0.5 psid (Refs. 1 and 2). Additionally, 1 of the 5 vacuum breakers is assumed to fail in a closed position (Ref. 1). The results of the analyses show that the design differential pressure is not exceeded even under the worst case accident scenario. The vacuum breaker opening differential pressure setpoint and the requirement that all vacuum breakers be OPERABLE (the additional vacuum breaker is required to meet the single failure criterion) are a result of the requirement placed on the vacuum breakers to limit the vent system waterleg height. The cross sectional areas of the vacuum breakers are sized on the basis of the Bodega Bay pressure suppression system tests. The vacuum breaker capacity selected on this test basis is more than adequate to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to a value which is within the suppression system design values (Refs. 3 and 4). Design Basis Accident (DBA)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

analyses assume the vacuum breakers to be closed initially and to remain closed and leak tight, until the suppression pool is at a positive pressure relative to the drywell.

The suppression chamber-to-drywell vacuum breakers satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

LCO

All vacuum breakers must be OPERABLE for opening. All suppression chamber-to-drywell vacuum breakers are also required to be closed (except during testing or when the vacuum breakers are performing their intended design function). The vacuum breaker OPERABILITY requirement provides assurance that the drywell-to-suppression chamber negative differential pressure remains below the design value. The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall, caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture that purges the drywell of air and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3. Excessive negative pressure inside the drywell could also occur due to inadvertent actuation of the RHR Containment Spray System during normal operation.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES; therefore, maintaining suppression chamber-to-drywell vacuum breakers OPERABLE is not required in MODE 4 or 5.

(continued)

BASES (continued)

ACTIONS

A.1

With one of the vacuum breakers inoperable for opening (e.g., the vacuum breaker is not open and may be stuck closed or not within its opening setpoint limit, so that it would not function as designed during an event that depressurized the drywell), the remaining four OPERABLE vacuum breakers are capable of providing the vacuum relief function. However, overall system reliability is reduced because a single active failure in one of the remaining vacuum breakers could result in an excessive negative drywell-to-suppression chamber differential pressure during a DBA. Therefore, with one of the five vacuum breakers inoperable, 72 hours is allowed to restore the inoperable vacuum breaker to OPERABLE status so that plant conditions are consistent with those assumed for the design basis analysis. The 72 hour Completion Time is considered acceptable due to the low probability of an event occurring that would require the remaining vacuum breaker capability.

B.1

An open vacuum breaker allows communication between the drywell and suppression chamber airspace, and, as a result, there is the potential for primary containment overpressurization due to bypass leakage if a LOCA were to occur. Therefore, the open vacuum breaker must be closed. A short time is allowed to close the vacuum breaker due to the low probability of an event that would pressurize primary containment. If vacuum breaker position indication is not reliable, an alternate method of verifying that the vacuum breakers are closed is to verify the bypass leakage between the drywell and suppression chamber is within the limits of SR 3.6.1.1.2 or by local observation. The required 2 hour Completion Time is considered adequate to perform this test. If the leak test fails, not only must this ACTION be taken (close the open vacuum breaker within the required Completion Time), but also the appropriate Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," must be entered.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.1

Each vacuum breaker is verified closed to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing local or relay room vacuum breaker position indication or by performing SR 3.6.1.1.2, the bypass leakage test. If the bypass test fails, not only must the vacuum breaker(s) be considered open and the appropriate Conditions and Required Actions of this LCO be entered, but also the appropriate Condition and Required Actions of LCO 3.6.1.1 must be entered. Each suppression chamber-to-drywell vacuum breaker disc will be seated as long as the arm movement is ≤ 1.0 degree. The vacuum breakers are considered closed if the associated position light indicates the closed position since it is set to actuate at ≤ 1.0 degree. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience. J

Two Notes are added to this SR. The first Note allows suppression chamber-to-drywell vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers.

The second Note is included to clarify that vacuum breakers open due to an actual differential pressure are not considered as failing this SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.7.2

Each vacuum breaker must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This ensures that the safety analysis assumptions are valid. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.1.7.3

Verification of the vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker opening differential pressure of 0.5 psid is valid. The 24 month Frequency has been shown to be acceptable, based on operating experience, and is further justified because SR 3.6.1.7.2 is performed at a shorter Frequency that conveys the proper functioning status of each vacuum breaker.

REFERENCES

1. UFSAR, Section 14.6.1.3.3.
2. UFSAR, Section 5.2.3.6.
3. UFSAR, Section 5.2.4.2.
4. Preliminary Hazards Summary Report, Bodega Bay Atomic Park Unit Number 1, Docket No. 50-205, Appendix I, December 28, 1962.
5. 10 CFR 50.36(c)(2)(ii).

DBI

INSERT BKG-1

Each subsystem collects leakage from the stem packing of all four outboard main steam isolation valves (MSIVs) and downstream of all outboard MSIVs. Each subsystem consists of valves, controls and piping which can be aligned to the Standby Gas Treatment (SGT) System for processing. During operation, the SGT System maintains sufficient negative pressure to provide the MSLC System flow required to ensure that all postulated leakage is collected and processed (Ref. 1). While both the stem packing and the downstream portion of each subsystem contribute to reducing uncontrolled or untreated MSIV leakage, the downstream portion performs the primary function of the MSLC System to collect and process the leakage across the MSIV seats. The downstream portion is provided with interlocks that prevent inadvertent operation of the system during normal operation and to prevent improper system lineup during accident conditions.

Each downstream portion of the MSLC subsystems includes a remote manual isolation valve, an automatic isolation valve, and a backup automatic isolation valve. A pressure switch which monitors MSLC System piping pressure is provided for each automatic isolation valve. These pressure switches act to prevent the opening of the valves and to automatically close the valves on high pressure. The pressure switches will indicate low pressure during normal plant operation since the remote manual isolation valves will isolate the pressure switches from main steam pressure. The operator initiates the operation of the stem packing portion of the MSLC subsystem by opening the associated remote manual isolation valve. The operator initiates operation of the downstream portion of each MSLC subsystem by first opening the associated remote manual isolation valve. The operator then places the control switch associated with the automatic isolation valves to open. If the MSLC System pressure is greater than 16 psig the valves will remain shut and automatically open at or below 16 psig.

PA1 PA2

8 2

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.~~9.3~~ (continued)

DB5

~~[18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~

While this Surveillance can be performed with the reactor at power,

DB5

Operating experience has shown that these components usually pass the Surveillance when performed at the ~~[18]~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

X2

REFERENCES

PA2

1. 4 FSAR, Section ~~[6.5]~~ 9.19.

DB4

2. Regulatory Guide 1.96, Revision ~~[1]~~, "Design of Main Steam Isolation Valve Leakage Control Systems For Boiling Water Reactor Nuclear Power Plants," June 1976

PA3

3. 10 CFR 50.36 (c)(2)(ii).

X1

J

Revision J

3.6 CONTAINMENT SYSTEMS

3.6.1.8 Main Steam Leakage Collection (MSLC) System

LCO 3.6.1.8 Two MSLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One MSLC subsystem inoperable.	A.1	Restore MSLC subsystem to OPERABLE status.	30 days
B. Two MSLC subsystems inoperable.	B.1	Restore one MSLC subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1 AND C.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.8 Main Steam Leakage Collection (MSLC) System

BASES

BACKGROUND

The MSLC System supplements the isolation function of the MSIVs by processing the fission products that could leak through the closed MSIVs after a Design Basis Accident (DBA) loss of coolant accident (LOCA).

The MSLC System consists of two independent and redundant subsystems. Each subsystem collects leakage from the stem packing of all four outboard main steam isolation valves (MSIVs) and downstream of all outboard MSIVs. Each subsystem consists of valves, controls and piping which can be aligned to the Standby Gas Treatment (SGT) System for processing. During operation, the SGT System maintains sufficient negative pressure to provide the MSLC System flow required to ensure that all postulated leakage is collected and processed (Ref. 1). While both the stem packing and the downstream portion of each subsystem contribute to reducing uncontrolled or untreated MSIV leakage, the downstream portion performs the primary function of the MSLC System to collect and process the leakage across the MSIV seats. The downstream portion is provided with interlocks that prevent inadvertent operation of the system during normal operation and to prevent improper system lineup during accident conditions.

Each downstream portion of the MSLC subsystems includes a remote manual isolation valve, an automatic isolation valve, and a backup automatic isolation valve. A pressure switch which monitors MSLC System piping pressure is provided for each automatic isolation valve. These pressure switches act to prevent the opening of the valves and to automatically close the valves on high pressure. The pressure switches will indicate low pressure during normal plant operation since the remote manual isolation valves will isolate the pressure switches from main steam pressure. The operator initiates the operation of the stem packing portion of the MSLC subsystem by opening the associated remote manual isolation valve. The operator initiates operation of the downstream portion of each MSLC subsystem by first opening the associated remote manual isolation valve. The operator

(continued)

BASES

BACKGROUND (continued)	then places the control switch associated with the automatic isolation valves to open. If the MSLC System pressure is greater than 16 psig the valves will remain shut and automatically open at or below 16 psig. 1B The MSLC System is manually initiated approximately 20 minutes following a DBA LOCA (Ref. 2).
APPLICABLE SAFETY ANALYSES	The MSLC System mitigates the consequences of a DBA LOCA by ensuring that fission products that may leak from the closed MSIVs are diverted to and filtered by the SGT System. The operation of the MSLC System prevents a release of untreated leakage for this type of event. The MSLC System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).
LCO	One MSLC subsystem can provide the required processing of the MSIV leakage. To ensure that this capability is available, assuming worst case single failure, two MSLC subsystems must be OPERABLE.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment. Therefore, MSLC System OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the MSLC System OPERABLE is not required in MODE 4 or 5 to ensure MSIV leakage is processed.
ACTIONS	<u>A.1</u> With one MSLC subsystem inoperable, the inoperable MSLC subsystem must be restored to OPERABLE status within

(continued)

<Insert 3.6.1.9> CLB1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.~~7~~.1 9 CLB1 [CTS 4.5.8.1.e]	~~-----NOTE----- RHR containment spray subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below [the RHR cut in permissive pressure in MODE 3] if capable of being manually realigned and not otherwise inoperable.~~ PA2 Verify each RHR containment spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position. PA1 1	31 days
SR 3.6.1.~~7~~.2 9 CLB1 [CTS 4.5.8.1.a]	Verify each required RHR pump develops a flow rate of ≥ ~~[5850]~~ 7750 gpm ~~on recirculation flow~~ while operating in cooling mode through the associated heat exchanger ~~to~~ the suppression pool. PA1 DB2 PA3	[In accordance with the Inservice Testing Program ~~or 92 days~~] CLB2 3
~~SR 3.6.1.7.3~~	~~Verify each RHR containment spray subsystem automatic valve in the flow path actuates to its correct position on an actual or simulated automatic initiation signal.~~	~~[18] months~~ DB1
SR 3.6.1.~~7~~.~~4~~ 3 9 CLB1 [CTS 4.5.8.1.f]	Verify each spray nozzle is unobstructed.	~~[At first refueling AND]~~ X1 10 years

Insert Page 3.6-30b

Bourson J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1434, REVISION 1
ITS: 3.6.1.9 - RESIDUAL HEAT REMOVAL (RHR) CONTAINMENT SPRAY

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 This Specification has been added in accordance with the current requirements in CTS 3.5.B.1. At JAFNPP both the drywell and suppression chamber sprays are required to mitigate the consequences of accidents. The current requirements are more consistent with Specification 3.6.1.7 of the BWR/6 Standard Technical Specifications, NUREG-1434, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)), therefore this Specification and Bases have been used to develop the ITS requirements of containment spray for the JAFNPP ITS submittal. The NUREG-1434 Specification and Surveillances have been renumbered as applicable.

CLB2 The brackets have been removed and the proper plant specific Surveillance Frequency has been included in accordance with CTS 4.5.B.1.a.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The word "required" has been included in ITS SR 3.6.1.9.2 in accordance with the use of this term in the Improved Technical Specifications. All RHR pumps are not required to be Operable to satisfy this Specification therefore this change is appropriate.

PA2 The Note to NUREG-1434, SR 3.6.1.7.1 is for BWR/6 plants where the RHR Containment Spray System is automatically initiated. The note has been deleted in the NUREG markup for ITS SR 3.6.1.9.1 because the RHR Containment Spray System at the FitzPatrick plant is manually initiated. The phrase "or can be aligned to the correct position" has been added to ITS SR 3.6.1.9.1 to be consistent with the format of the SRs of other manually initiated systems such as those addressed by NUREG-1433, SR 3.6.2.4.1 and NUREG-1433, SR 3.6.2.3.1 (ITS SR 3.6.2.3.1).

PA3 The term "on recirculation flow" in ITS SR 3.6.1.9.2 has been changed to state "while operating in the suppression pool cooling mode", consistent with the Bases description in the NUREG and with ISTS SR 3.6.2.3.2, a similar Surveillance Requirement. J

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 NUREG-1434 SR 3.6.1.7.3 has been deleted since it is not applicable. The JAFNPP design does not include any automatic actuation of the containment spray mode therefore this surveillance is not necessary.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1434, REVISION 1
ITS: 3.6.1.9 - RESIDUAL HEAT REMOVAL (RHR) CONTAINMENT SPRAY

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB2 The brackets have been removed and the proper plant specific value has been included.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 The bracketed surveillance Frequency in NUREG-1434 SR 3.6.1.7.4 (At first refueling) has been deleted since the first refueling outage is already completed. This surveillance was intended for new plants licensed under NUREG-1434.

<Insert B 3.6.1.9> CLB1

CLB1

BASES

APPLICABLE SAFETY ANALYSES (continued)

DB2

with containment spray operation the primary containment pressure remains within design limits.

X1

INSERT ASA

The RHR Containment Spray System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 5).

5

LCO

In the event of a Design Basis Accident (DBA), a minimum of one RHR containment spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure and temperature below design limits. To ensure that these requirements are met, two RHR containment spray subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR containment spray subsystem is OPERABLE when one of the pump, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

DB2 5 DB6

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause pressurization and heating of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining RHR containment spray subsystems OPERABLE is not required in MODE 4 or 5.

DB2

ACTIONS

A.1

With one RHR containment spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE RHR containment spray subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single active failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time was chosen in light of the redundant RHR containment capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

PA2

(continued)

Insert Page B 3.6-57c

Revision I

DB2

INSERT ASA

Steam line breaks have been analyzed to develop a drywell air temperature history for use in equipment qualification (Refs. 3 and 4). The RHR containment sprays are assumed to be initiated at a minimum time of 10 minutes. The RHR containment spray flow rates were assumed to be 7,150 gpm for drywell sprays and 600 gpm for suppression chamber sprays. The highest air temperature envelope is 335°F for the first 300 seconds and this is as a result of a 0.75 ft^2 steam line break (Ref. 4). The analysis (Ref. 4) concluded containment design temperature is not exceeded since drywell spray activation will terminate any further rise in drywell air temperature.

<Insert B3.6.1.9> CLB1

CLB1

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.6.1.7.1 (continued)~~

~~A Note has been added to this SR that allows RHR containment spray subsystems to be considered OPERABLE during alignment to and operation in the RHR shutdown cooling mode when below [the RHR cut in permissive pressure in MODE 3], if capable of being manually realigned and not otherwise inoperable. At these low pressures and decay heat levels (the reactor is shut down in MODE 3), a reduced complement of subsystems can provide the required containment pressure mitigation function thereby allowing operation of an RHR shutdown cooling loop when necessary.~~ PA3

CLB1 9

SR 3.6.1.~~7~~9.2

Verifying each required RHR pump develops a flow rate ≥ 7750 ~~[5650]~~ gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. It is tested in the pool cooling mode to demonstrate pump OPERABILITY without spraying down equipment in the drywell ~~primary containment~~. Flow is a normal test of centrifugal pump performance required by the ASME Code, Section XI (Ref. 6 ~~8~~). This test confirms one point on the pump performance ~~design~~ curve and is indicative of overall performance. Such inservice tests ~~inspections~~ confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. {The Frequency of this SR is in accordance with the Inservice Testing Program ~~or 92 days~~.} CLB2

PA1 DB4 PA4 DB2 J PA2

~~SR 3.6.1.7.3~~

~~This SR verifies that each RHR containment spray subsystem automatic valve actuates to its correct position upon receipt of an actual or simulated automatic actuation signal. Actual spray initiation is not required to meet this SR. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.3.6 overlaps this SR to provide complete testing of the safety function. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at~~ DB3

(continued)

Insert Page B3.6-57.f Revision J

<Insert B 3.6.1.9> CLB1

CLB1

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.6.1.7.3 (continued)~~

~~the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ DB3

CLB1 DB3

SR 3.6.1.~~7.4~~ 9 3

This Surveillance is performed every 10 years by introduction of air PA5 to verify that the spray nozzles are not obstructed and that flow will be provided when required. The 10 year Frequency is adequate to detect degradation in performance due to the passive nozzle design and its normally dry state and has been shown to be acceptable through operating experience.

PA4

REFERENCES

DB2

Insert Ref-1

2 ~~1~~. 4 FSAR, Section ~~6.2.1.1.5~~ 5.2.4.4. DB5

DB2

~~2~~ 5. ASME, Boiler and Pressure Vessel Code, Section XI.

DB2

Insert Ref-2

DB2

X1

Insert Page B3.6-57g

Revision J

<Insert B 3.6.1.9> CCB1

DB2 INSERT Ref-1

1. UFSAR, Table 5.2-1.

X1 DB2 INSERT Ref-2

3. UFSAR, Section 14.6.

4. GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, August 1996. J

5. 10 CFR 50.36(c)(2)(ii). J

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.9.1	Verify each RHR containment spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.1.9.2	Verify each required RHR pump develops a flow rate of ≥ 7750 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	In accordance with the Inservice Testing Program
SR 3.6.1.9.3	Verify each spray nozzle is unobstructed.	10 years

J

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum allowable equivalent flow path area for bypass leakage has been specified to be 0.032 ft^2. The analysis demonstrates that with containment spray operation the primary containment pressure remains within design limits.

Steam line breaks have been analyzed to develop a drywell air temperature history for use in equipment qualification (Refs. 3 and 4). The RHR containment sprays are assumed to be initiated at a minimum time of 10 minutes. The RHR containment spray flow rates were assumed to be 7,150 gpm for drywell sprays and 600 gpm for suppression chamber sprays. The highest air temperature envelope is 335°F for the first 300 seconds and this is as a result of a 0.75 ft^2 steam line break (Ref. 4). The analysis (Ref. 4) concluded containment design temperature is not exceeded since drywell spray activation will terminate any further rise in drywell air temperature.

The RHR Containment Spray System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

LCO

In the event of a Design Basis Accident (DBA), a minimum of one RHR containment spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure and temperature below design limits. To ensure that these requirements are met, two RHR containment spray subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR containment spray subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause pressurization and heating of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining RHR containment spray subsystems OPERABLE is not required in MODE 4 or 5.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.9.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR containment spray mode flow path provides assurance that the proper flow paths will exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR Containment Spray System is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR is justified because the valves are operated under procedural control and because improper valve position would affect only a single subsystem. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.1.9.2

Verifying each required RHR pump develops a flow rate ≥ 7750 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. It is tested in the pool cooling mode to demonstrate pump OPERABILITY without spraying down equipment in the drywell. Flow is a normal test of centrifugal pump performance required by the ASME Code, Section XI (Ref. 6). This test confirms one point on the pump performance curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.9.3

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

REFERENCES

1. UFSAR, Table 5.2-1.
2. UFSAR, Section 5.2.4.4.
3. UFSAR, Section 14.6.
4. GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, August 1996.
5. 10 CFR 50.36(c)(2)(ii).
6. ASME, Boiler and Pressure Vessel Code, Section XI.

3.6 CONTAINMENT SYSTEMS

3.6.2.1 Suppression Pool Average Temperature

[3.7.A.1]

LCO 3.6.2.1 Suppression pool average temperature shall be:

DB1

[3.7.A.1.c(1)]

with THERMAL POWER > 1% RTP

a. ≤ [95]°F ~~when any OPERABLE intermediate range monitor (IRM) channel is > [25/40] divisions of full scale on Range 7~~ and no testing that adds heat to the suppression pool is being performed; DB1 1 (J)

[3.7.A.1.c(2)]

CLB1 TA1

b. ≤ [105]°F ~~when any OPERABLE IRM channel is > [25/40] divisions of full scale on Range 7~~ and testing that adds heat to the suppression pool is being performed; and

[3.7.A.1.c(2)]

with THERMAL POWER ≤ 1% RTP

DB1

c. ≤ [110]°F ~~when all OPERABLE IRM channels are ≤ [25/40] divisions of full scale on Range 7.~~ 1 (J)

[3.7.A.1]
[M4]

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

[3.7.A.1.c(2)]
[M1]
[L1]

TA1 CLB1

THERMAL POWER > 1% RTP.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool average temperature > [95]°F but ≤ [110]°F. DB1 AND ~~Any OPERABLE IRM channel > [25/40] divisions of full scale on Range 7.~~ AND Not performing testing that adds heat to the suppression pool.	A.1 Verify suppression pool average temperature ≤ [110]°F. DB1 AND A.2 Restore suppression pool average temperature to ≤ [95]°F. DB1	Once per hour 24 hours

(continued)

Revision J

DB2 were cases addressed as part of the pool temperature analyses of Reference 2 J

J

BASES (continued)

APPLICABLE SAFETY ANALYSES

Insert ASA-1

DB2

DB1

Insert ASA-2

The postulated DBA against which the primary containment performance is evaluated is the entire spectrum of postulated pipe breaks within the primary containment. Inputs to the safety analyses include initial ~~suppression pool water volume and~~ suppression pool temperature (Reference 1 for LOCAs and ~~Reference 2 for the pool temperature analyses required by Reference 3~~). DB2 J An initial pool temperature of [95]°F is assumed for the Reference 1 ~~and Reference 2~~ 5 2, and 3 analyses. J Reactor shutdown at a pool temperature of [110]°F and vessel depressurization at a pool temperature of [120]°F ~~are assumed for the Reference 2 analyses~~. J The limit of [105]°F, at which testing is terminated, is not used in the safety analyses because DBAs are assumed to not initiate during ~~unit~~ plant PA2 testing. J

Suppression pool average temperature satisfies Criteria 2 and 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 5). X1 J

LCO

TA1 CLB1 PA4

with THERMAL POWER > 1% RTP

with THERMAL POWER > 1% RTP

TA1 CLB1

plant PA2

A limitation on the suppression pool average temperature is required to provide assurance that the containment conditions assumed for the safety analyses are met. This limitation ~~subsequently~~ ensures that peak primary containment pressures and temperatures do not exceed maximum allowable values during a postulated DBA or any transient resulting in heatup of the suppression pool. The LCO requirements are: J

DB1

a. Average temperature ≤ [95]°F ~~when any OPERABLE intermediate range monitor (IRM) channel is > [25/40] divisions of full scale on Range 7~~ and no testing that adds heat to the suppression pool is being performed. This requirement ensures that licensing bases initial conditions are met.

DB1

b. Average temperature ≤ [105]°F ~~when any OPERABLE IRM channel is > [25/40] divisions of full scale on Range 7~~ and testing that adds heat to the suppression pool is being performed. This required value ensures that the ~~unit~~ has testing flexibility, and was selected to provide margin below the [110]°F limit at which reactor shutdown is required. When testing ends, temperature must be restored to ≤ [95]°F within 24 hours according to Required Action A.2. Therefore, the time period that the temperature is > [95]°F is DB1

(continued)

Revision J

DB2

INSERT ASA-1

References 2 and 3 for the pool temperature analyses required by Reference 4

DB2

INSERT ASA-2

The limiting case of rapid depressurization from isolated Hot Shutdown (reactor scram and main steam isolation valve closure, with initial pool temperature of 95°F) with assumed loss of one residual heat removal loop (Reference 2) was addressed as part of the analyses of Reference 3.

PA2

BASES

LCO (continued)

plant

PA2

short enough not to cause a significant increase in ~~unit~~ risk.

DB1

plant

c. Average temperature ≤ [110]°F ~~when all OPERABLE IRM channels are ≤ [25/40] divisions of full scale on Range 7~~. This requirement ensures that the ~~unit~~ will be shut down at > [110]°F. The pool is designed to absorb decay heat and sensible heat but could be heated beyond design limits by the steam generated if the reactor is not shut down.

CLB1

TA1

with THERMAL POWER ≥ 1% RTP

PB1

CLB1

TA1

1% RTP

Insert LCO-1

PA6

At this condition

~~Note that~~ [25/40] divisions of full scale on IRM Range 7 is a convenient measure of ~~when the~~ reactor ~~is producing~~ power essentially equivalent to 1% RTP. At ~~this power level~~, heat input is approximately equal to normal system heat losses.

Insert LCO-2

PA6

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant heatup of the suppression pool. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining suppression pool average temperature within limits is not required in MODE 4 or 5.

ACTIONS

<u>A.1 and A.2</u>

DB2

With the suppression pool average temperature above the specified limit when not performing testing that adds heat to the suppression pool and when above the specified power indication, the initial conditions exceed the conditions assumed for the Reference 1, ~~3~~, and ~~4~~ analyses. However, primary containment cooling capability still exists, and the primary containment pressure suppression function will occur at temperatures well above those assumed for safety analyses. Therefore, continued operation is allowed for a limited time. The 24 hour Completion Time is adequate to allow the suppression pool average temperature to be restored below the limit. Additionally, when suppression pool temperature is > [95]°F, increased monitoring of the suppression pool temperature is required to ensure that it remains ≤ [110]°F. The once per hour Completion Time is adequate based on past experience, which has shown that pool temperature increases relatively slowly except when testing

DB1

(continued)

Revision J

BASES

ACTIONS

D.1 and D.2 (continued) D.3 PA3

experience. Given the high suppression pool average temperature in this Condition, the monitoring Frequency is increased to twice that of Condition A. Furthermore, the 30 minute Completion Time is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

E.1 and E.2

If suppression pool average temperature cannot be maintained at ≤ [120]°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor pressure must be reduced to < [200] psig within 12 hours, and the plant must be brought to at least MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. DB1

Continued addition of heat to the suppression pool with suppression pool temperature > [120]°F could result in exceeding the design basis maximum allowable values for primary containment temperature or pressure. Furthermore, if a blowdown were to occur when the temperature was > [120]°F, the maximum allowable bulk and local temperatures could be exceeded very quickly. DB1

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1

The suppression pool average temperature is regularly monitored to ensure that the required limits are satisfied. ~~The average temperature is determined by taking an arithmetic average of OPERABLE suppression pool water temperature channels.~~ The 24 hour Frequency has been shown, based on operating experience, to be acceptable. When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute Frequency during testing is justified by the rates at which tests will heat up the suppression pool, has been shown to be acceptable based on

The LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," Bases contains a description of the suppression pool temperature monitoring system DB3

1J

1J

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1 (continued)

operating experience, and provides assurance that allowable pool temperatures are not exceeded. The Frequencies are further justified in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

REFERENCES

PA2 ④ DB4

1. FSAR, Section ~~[6.2]~~ 14.6.1.3.3.

2. FSAR, Section [15.1].

3. NUREG-0783.

4. [Mark I Containment Program].

GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, August 1996.

NEDC-24361-P, James A. FitzPatrick Nuclear Power Plant Suppression Pool Temperature Response, August 1981.

5. 10 CFR 50.36 (c) (2) (ii).

Letter from R. W. Reid (NRC) to G. T. Berry (NYPA), Request for Additional Information Regarding Suppression Pool Temperature Transients, December 9, 1977

DB2 DB2 X1 DB4

J J J J J J J

Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 THERMAL POWER in the range of 1% RTP is not readily quantified with much accuracy. While range 7 on the IRMs approximates 1% RTP, this power level can also be approximated from SRMs and even by determining the point of adding heat. These acceptable options are desired to be maintained in plant procedures, with the ITS requirement as it is in the JAFNPP Technical Specifications; i.e., 1% RTP (in accordance with the definition of reactor power operation). Therefore, the LCO and ACTIONS have been modified to reflect the 1% RTP requirement. The changes marked "CLB1" use words and phrases that are identical to those used in TSTF-206, R0, and are also marked "TA1." See Bases JFD TA1 below.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The bracketed discussions of the four different concerns that lead to the development of the suppression pool average temperature limits have been deleted. The discussion in the proposed Bases provides sufficient information to understand this Specification.

PA2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA3 The Bases have been revised to be consistent with the Specifications.

PA4 A typographical or editorial error has been corrected.

PA5 Not used. 1 J

PA6 Changes have been made to provide more detailed description of the methods that can be used to determine whether the plant is operating at 1% RTP.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific value has been provided.

DB2 The Bases have been revised to reflect the JAFNPP specific references.

3.6 CONTAINMENT SYSTEMS

3.6.2.1 Suppression Pool Average Temperature

LCO 3.6.2.1 Suppression pool average temperature shall be:

a. ≤ 95°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 > 95°F but ≤ 110°F. <u>AND</u> THERMAL POWER > 1% RTP. <u>AND</u> Not performing testing that adds heat to the suppression pool.	A.1 Verify suppression pool average temperature ≤ 110°F. <u>AND</u> A.2 Restore suppression pool average temperature to ≤ 95°F.	Once per hour 24 hours

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.1 Suppression Pool Average Temperature

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown from safety/relief valve discharges or from Design Basis Accidents (DBAs). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment that ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from steam exhaust lines in the turbine driven systems (i.e., the High Pressure Coolant Injection System and Reactor Core Isolation Cooling System). Suppression pool average temperature (along with LCO 3.6.2.2, "Suppression Pool Water Level") is a key indication of the capacity of the suppression pool to fulfill these requirements. 1/J

The technical concerns that lead to the development of suppression pool average temperature limits are as follows:

a. Complete steam condensation;

b. Primary containment peak pressure and temperature;

c. Condensation oscillation loads; and

d. Chugging loads.

APPLICABLE SAFETY ANALYSES

The postulated DBA against which the primary containment performance is evaluated is the entire spectrum of postulated pipe breaks within the primary containment. Inputs to the safety analyses include initial suppression pool temperature (Reference 1 for LOCAs and References 2 and 3 for the pool temperature analyses required by Reference 4). An initial pool temperature of 95°F is assumed for the References 1, 2, and 3 analyses. Reactor shutdown at a pool temperature of 110°F and vessel depressurization at a pool temperature of 120°F were cases addressed as part of the J 1/I

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	pool temperature analyses of Reference 2. The limiting case of rapid depressurization from isolated Hot Shutdown (reactor scram and main steam isolation valve closure, with initial pool temperature of 95°F) with assumed loss of one residual heat removal loop (Reference 2) was addressed as part of the analyses of Reference 3. The limit of 105°F, at which testing is terminated, is not used in the safety analyses because DBAs are assumed to not initiate during plant testing. 5 Suppression pool average temperature satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5). 5
LCO	A limitation on the suppression pool average temperature is required to provide assurance that the containment conditions assumed for the safety analyses are met. This limitation ensures that peak primary containment pressures and temperatures do not exceed maximum allowable values during a postulated DBA or any transient resulting in heatup of the suppression pool. The LCO requirements are: a. Average temperature ≤ 95°F with THERMAL POWER > 1% RTP and no testing that adds heat to the suppression pool is being performed. This requirement ensures that licensing bases initial conditions are met. b. Average temperature ≤ 105°F with THERMAL POWER > 1% RTP and testing that adds heat to the suppression pool is being performed. This required value ensures that the plant has testing flexibility, and was selected to provide margin below the 110°F limit at which reactor shutdown is required. When testing ends, temperature must be restored to ≤ 95°F within 24 hours according to Required Action A.2. Therefore, the time period that the temperature is > 95°F is short enough not to cause a significant increase in plant risk.

(continued)

BASES

LCO (continued)

c. Average temperature ≤ 110°F with THERMAL POWER ≤ 1% RTP. This requirement ensures that the plant will be shut down at > 110°F. The pool is designed to absorb decay heat and sensible heat but could be heated beyond design limits by the steam generated if the reactor is not shut down.

Indication of 1% RTP varies with plant conditions and can be determined by more than one method. When at or near normal operating temperature, Reactor Coolant System (RCS) losses such as the Reactor Water Cleanup System, steam line drains and insulation inefficiency are approximately 1% RTP or less and reactor power level can be observed on the intermediate range monitor (IRM) Instrumentation. At this condition 25/40 divisions of full scale on IRM Range 7 is a convenient measure of reactor power essentially equivalent to 1% RTP. At 1% RTP, heat input is approximately equal to normal system heat losses. When RCS temperature is significantly below the normal operating temperature, maintaining reactor power level at or below the "point of adding heat" maintains power level well below 1% RTP.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant heatup of the suppression pool. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining suppression pool average temperature within limits is not required in MODE 4 or 5.

ACTIONS

<u>A.1 and A.2</u>

With the suppression pool average temperature above the specified limit when not performing testing that adds heat to the suppression pool and when above the specified power indication, the initial conditions exceed the conditions assumed for the References 1, 2, and 3 analyses. However, primary containment cooling capability still exists, and the primary containment pressure suppression function will occur at temperatures well above those assumed for safety analyses. Therefore, continued operation is allowed for a limited time. The 24 hour Completion Time is adequate to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1 (continued)

The LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," Bases contains a description of the suppression pool temperature monitoring system. The 24 hour Frequency has been shown, based on operating experience, to be acceptable. When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute Frequency during testing is justified by the rates at which tests will heat up the suppression pool, has been shown to be acceptable based on operating experience, and provides assurance that allowable pool temperatures are not exceeded. The Frequencies are further justified in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

REFERENCES

1. UFSAR, Section 14.6.1.3.3.

2. NEDC-24361-P, James A. FitzPatrick Nuclear Power Plant Suppression Pool Temperature Response, August 1981.

3. GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, August 1996.

4. Letter from R. W. Reid (NRC) to G. T. Berry (NYPA), Request for Additional Information Regarding Suppression Pool Temperature Transients, December 9, 1977.

5. 10 CFR 50.36(c)(2)(ii).

Specification 3.6.2.2

A1

3.7 LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

[3.6.2.2] A. Primary Containment — Suppression Pool Water Level

See ITS: 3.6.2.1

[LCO 3.6.2.2] 1. The level from the bottom of the torus and temperature of the water in the torus shall be maintained within the following limits whenever the reactor is critical or whenever the reactor coolant temperature is greater than 212°F and irradiated fuel is in the reactor vessel:

[Applicability MODES 1, 2, and 3]

M2

[LCO 3.6.2.2]

a. Maximum level of 14.00 feet.

b. Minimum level of 13.88 feet.

add Required Action A.1

L1

[Note to LCO 3.6.2.2]

The torus water level may be outside the above limits for a maximum of four (4) hours as a result of required operability testing of HPCI, RCIC, RHR, CS, and the Drywell - Torus Vacuum Relief System.

LA1

c. Maximum water temperature

(1) During normal power operation maximum water temperature shall be 95°F.

See ITS: 3.6.2.1

4.7 SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

see ITS: 3.6.2.1

[SR 3.6.2.2.1] 1. The torus water level and temperature shall be monitored as specified in Table 4.2-8. every 24 hours

A2

See ITS: 3.6.1.1

The accessible interior surfaces of the drywell and above the water line of the torus shall be inspected once per 24 months for evidence of deterioration.

Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continuously recorded until the heat addition is terminated. The operator will verify that average temperature is within applicable limits every 5 minutes. In lieu of continuous recording, the operator shall log the temperature every 5 minutes.

Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the torus shall be conducted before resuming power operation.

see ITS: 3.6.2.1

J

TECHNICAL CHANGES - MORE RESTRICTIVE

M2 (continued)

level will be required to be Operable at all times in MODE 2 even prior to any plant startup when reactor coolant temperature may be below 212°F. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS 3.7.A.1.b allows the torus (suppression pool) water level to be outside the limits for a maximum of 4 hours as a result of required operability testing of HPCI, RCIC, RHR, CS, and the Drywell-Torus Vacuum Relief System. The details of which Surveillances this allowance is provided for is proposed to be relocated to the Bases. The allowance in the Note to ITS LCO 3.6.2.2 that the limit is not required to be met for 4 hours during Surveillances that cause the suppression pool water level to be outside the limit is adequate to ensure the allowance is taken only during planned testing. The specific details of the Operability Note is not necessary to be in the Specification. As such, these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 ITS 3.6.2.2 ACTION A has been added to CTS 3.7.A.1 for suppression pool water level outside of limits. Currently, no time is allowed to restore level unless required operability testing is being performed (CTS 3.7.A.1). An unanticipated change in the suppression pool level would require addressing the cause and aligning the appropriate system to raise or lower the pool level. These activities require some time to accomplish. The Completion Time of 2 hours is based on engineering judgement of the relative risks associated with: 1) the safety significance; 2) the probability of an event requiring the safety function of the system; and 3) the relative risks associated with the plant transient and the potential challenge to safety systems experienced by requiring a plant shutdown. Upon further review and discussion with the NRC staff during the development of NUREG-1433, a 2 hour Completion Time was determined to be appropriate.

L2 CTS 3.7.A.8 requires the reactor to be in the cold condition within 24 hours if the requirements of Specification 3.7.A.1 cannot be met. ITS 3.6.2.2 ACTION A allows 2 hours to restore suppression pool water level

3.7.A.1
3.7.A.1.a
3.7.A.1.b

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Suppression pool water level shall be ≥ ~~[12]~~ 13.88 ft ~~[2 inches]~~ and ≤ ~~[12]~~ 14 ft ~~[6 inches]~~. DB1 5

3.7.A.1

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

L1

3.7.A.8 M1 L2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.2.2.1 Verify suppression pool water level is within limits.	24 hours

---- NOTE ----
Not required to be met for up to 4 hours during Surveillances that cause suppression pool water level to be outside the limit.

3.7.A.1.b

CLB1

5

Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.6.2.2 - SUPPRESSION POOL WATER LEVEL

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The Note to ITS LCO 3.6.2.2 has been added in accordance with the current allowances in CTS 3.7.A.1.b. This additional allowance is needed since the suppression pool level band is less than 2 inches. | J

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific value has been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

BASES (continued)

system downcomer PA3

APPLICABLE SAFETY ANALYSES

Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent clearing for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to S/RV discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of Reference 1 and 2 remain valid. DB4

Suppression pool water level satisfies Criteria 2 and 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii) (Ref. 3). X1

LCO

A limit that suppression pool water level be ≥ ~~[12]~~ 13.88 ft ~~[2 inches]~~ and ≤ ~~[12]~~ 14 ft ~~[6 inches]~~ is required to ensure that the primary containment conditions assumed for the safety analyses are met. Either the high or low water level limits were used in the safety analyses, depending upon which is more conservative for a particular calculation. DB1

CLB1 Insert LCO

APPLICABILITY

In MODES 1, 2, and 3, a DBA would cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. The requirement~~s~~ for maintaining suppression pool water level within limits in MODE 4 or 5 is addressed in LCO 3.5.2, "ECCS-Shutdown." PA2

ACTIONS

A.1

the vent system downcomer lines PA3

With suppression pool water level outside the limits, the conditions assumed for the safety analyses are not met. If water level is below the minimum level, the pressure suppression function still exists as long as ~~main vents~~ the vent system downcomer lines are covered, HPCI and RCIC turbine exhausts are covered, and S/RV quenchers are covered. If suppression pool water level is above the maximum level, protection against overpressurization still exists due to the margin in the peak containment pressure analysis and the capability of the ~~Drywell~~ Residual Heat Removal Containment Spray System. Therefore, continued operation for a

PA3

(continued)

Revision I

CLB1 INSERT LCO 3.6.2.2

The LCO is modified by a Note which states that the LCO is not required to be met for up to four hours during Surveillances that cause suppression pool water level to be outside of limits. These Survillances include required OPERABILITY testing of the High Pressure Coolant Injection System, the Reactor Core Isolation Cooling System, the suppression chamber-to-drywell vacuum breakers, the Core Spray System and the Residual Heat Removal System. The 4 hour allowance is adequate to perform the Surveillances and to restore the suppression pool water level to within limits.

BASES

ACTIONS

A.1 (continued)

requiring

limited time is allowed. The 2 hour Completion Time is sufficient to restore suppression pool water level to within limits. Also, it takes into account the low probability of an event impacting the suppression pool water level to be within limits occurring during this interval.

PA3

B.1 and B.2

If suppression pool water level cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2.1

Verification of the suppression pool water level is to ensure that the required limits are satisfied. ~~The 24 hour Frequency of this SR was developed considering operating experience related to trending variations in suppression pool water level and water level instrument drift during the applicable MODES and to assessing the proximity to the specified LCO level limits.~~ Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool water level condition.

DO2

The 24 hour Frequency has been shown to be acceptable based on operating experience.

REFERENCES

PA1 DB3

1. U FSAR, Section ~~6.2~~ 14.6.1.3.3.

2. GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, August 1996.

3. 10 CFR 50.36 (c)(2)(ii).

X1

Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.2.2 - SUPPRESSION POOL WATER LEVEL

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The Note to ITS LCO 3.6.2.2 has been added in accordance with the current allowances in CTS 3.7.A.1.b. This additional allowance is needed since the suppression pool level band is less than 2 inches. The Bases have been modified to reflect this change. | J

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA2 A typographical error has been corrected.

PA3 The Bases have been revised for enhanced clarity or to be consistent with other places in the Bases.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific value has been provided.

DB2 The Bases have been revised to more accurately reflect the basis for the 24 hour Frequency of SR 3.6.2.2.1.

DB3 The brackets have been removed and the proper plant specific reference has been provided.

DB4 Changes have been made (additions, deletions and/or changes) to reflect the plant specific Reference.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Suppression pool water level shall be ≥ 13.88 ft and ≤ 14 ft.

------------------------NOTE------------------------
Not required to be met for up to 4 hours during Surveillances that cause suppression pool water level to be outside the limit.

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. <u>AND</u> 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

BASES (continued)

APPLICABLE SAFETY ANALYSES

Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent system downcomer clearing for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to S/RV discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of References 1 and 2 remain valid.

Suppression pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

A limit that suppression pool water level be ≥ 13.88 ft and ≤ 14 ft is required to ensure that the primary containment conditions assumed for the safety analyses are met. Either the high or low water level limits were used in the safety analyses, depending upon which is more conservative for a particular calculation.

The LCO is modified by a note which states that the LCO is not required to be met up to four hours during Surveillances that cause suppression pool water level to be outside of limits. These Surveillances include required OPERABILITY testing of the High Pressure Coolant Injection System, the Reactor Core Isolation Cooling System, the suppression chamber-to-drywell vacuum breakers, the Core Spray System and the Residual Heat Removal System. The 4 hour allowance is adequate to perform the Surveillances and to restore the suppression pool water level to within limits.

APPLICABILITY

In MODES 1, 2, and 3, a DBA would cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. The requirement for maintaining suppression pool water level within limits in MODE 4 or 5 is addressed in LCO 3.5.2, "ECCS – Shutdown."

(continued)

BASES (continued)

ACTIONS

A.1

With suppression pool water level outside the limits, the conditions assumed for the safety analyses are not met. If water level is below the minimum level, the pressure suppression function still exists as long as the vent system downcomer lines are covered, HPCI and RCIC turbine exhausts are covered, and S/RV quenchers are covered. If suppression pool water level is above the maximum level, protection against overpressurization still exists due to the margin in the peak containment pressure analysis and the capability of the Residual Heat Removal Containment Spray System. Therefore, continued operation for a limited time is allowed. The 2 hour Completion Time is sufficient to restore suppression pool water level to within limits. Also, it takes into account the low probability of an event requiring the suppression pool water level to be within limits occurring during this interval.

B.1 and B.2

If suppression pool water level cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2.1

Verification of the suppression pool water level is to ensure that the required limits are satisfied. The 24 hour Frequency has been shown to be acceptable based on operating experience. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool water level condition.

J

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 14.6.1.3.3.

2. GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, August 1996. J

3. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

BASES

BACKGROUND

Following a Design Basis Accident (DBA), the RHR Suppression Pool Cooling System removes heat from the suppression pool. The suppression pool is designed to absorb the sudden input of heat from the primary system. In the long term, the pool continues to absorb residual heat generated by fuel in the reactor core. Some means must be provided to remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. This function is provided by two redundant RHR suppression pool cooling subsystems. The purpose of this LCO is to ensure that both subsystems are OPERABLE in applicable MODES.

(loop)

Each RHR suppression pool cooling ~~subsystem~~ contains two pumps and one heat exchanger and is manually initiated and independently controlled. The two subsystems perform the suppression pool cooling function by circulating water from the suppression pool through the RHR heat exchangers and returning it to the suppression pool. RHR service water, circulating through the tube side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the ~~external~~ ultimate heat sink.

PA2

PA1

The heat removal capability of one RHR pump ~~in one subsystem~~ is sufficient to meet the overall DBA pool cooling requirement for loss of coolant accidents (LOCAs) and transient events such as a turbine trip or stuck open safety/relief valve (S/RV). S/RV leakage, ~~and~~ high pressure ~~core~~ Coolant injection System and Reactor Core Isolation Cooling System testing increase suppression pool temperature more slowly. The RHR Suppression Pool Cooling System is also used to lower the suppression pool water bulk temperature following such events.

PA2

PA1

INSERT BKGD

DB1

APPLICABLE SAFETY ANALYSES

(s) (and 2)

DB1 J

Reference~~s~~ 1 and 2 contain~~s~~ the results of analyses used to predict primary containment pressure and temperature following large and small break LOCAs. The ~~intent of the~~ analyses ~~is to demonstrate~~ indicate that the heat removal capacity of the RHR Suppression Pool Cooling System is adequate to maintain the primary containment conditions within design limits. The

Insert ASA

DB1

(continued)

Revision J

DB1 INSERT BKGD

The RHR Suppression Pool Cooling System also ensures adequate net positive suction head (NPSH) is available for the Emergency Core Cooling System pumps.

DB1 INSERT ASA

References 2 and 3 contain the results of analyses used to predict local and bulk suppression pool temperatures following certain events including small break LOCAs and a stuck open S/RV. J

BASES

ACTIONS

A.1 (continued)

cooling capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

Insert ACTION B

TA1

C

B.1 and B.2

any

If ~~the~~ Required Action and associated Completion Time of Condition A cannot be met within the required Completion Time or if two RHR suppression pool cooling subsystems are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. 1/J

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR suppression pool cooling mode flow path provides assurance that the proper flow path exists for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is a manually initiated system. This Frequency PA3

(continued)

Revision J

INSERT REF

2. GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, August 1996.

3. NEDC-24361-P, James A. FitzPatrick Nuclear Power Plant Suppression Pool Temperature Response, August 1981.

4. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

BASES

BACKGROUND

Following a Design Basis Accident (DBA), the RHR Suppression Pool Cooling System removes heat from the suppression pool. The suppression pool is designed to absorb the sudden input of heat from the primary system. In the long term, the pool continues to absorb residual heat generated by fuel in the reactor core. Some means must be provided to remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. This function is provided by two redundant RHR suppression pool cooling subsystems. The purpose of this LCO is to ensure that both subsystems are OPERABLE in applicable MODES.

Each RHR suppression pool cooling subsystem (loop) contains two pumps and one heat exchanger and is manually initiated and independently controlled. The two subsystems perform the suppression pool cooling function by circulating water from the suppression pool through the RHR heat exchangers and returning it to the suppression pool. RHR service water, circulating through the tube side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the ultimate heat sink.

The heat removal capability of one RHR pump is sufficient to meet the overall DBA pool cooling requirement for loss of coolant accidents (LOCAs) and transient events such as a turbine trip or stuck open safety/relief valve (S/RV). S/RV leakage, High Pressure Coolant Injection System and Reactor Core Isolation Cooling System testing increase suppression pool temperature more slowly. The RHR Suppression Pool Cooling System is also used to lower the suppression pool water bulk temperature following such events. The RHR Suppression Pool Cooling System also ensures adequate net positive suction head (NPSH) is available for the Emergency Core Cooling System pumps.

APPLICABLE SAFETY ANALYSES

References 1 and 2 contain the results of analyses used to predict primary containment pressure and temperature following large and small break LOCAs. References 2 and 3 contain the results of analyses used to predict local and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	bulk suppression pool temperatures following certain events including small break LOCAs and a stuck open S/RV. The analyses indicate that the heat removal capacity of the RHR Suppression Pool Cooling System is adequate to maintain the primary containment conditions within design limits. The suppression pool temperature is calculated to remain below the design limit. The RHR Suppression Pool Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).
LCO	Following a DBA, a minimum of one RHR suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below design limits (Ref. 3). To ensure that these requirements are met, two RHR suppression pool cooling subsystems must be OPERABLE with power from two safety related redundant power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active component failure. An RHR suppression pool cooling subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment and cause a heatup and pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the RHR Suppression Pool Cooling System is not required to be OPERABLE in MODE 4 or 5.
ACTIONS	<u>A.1</u> With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1 (continued)

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

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

SR 3.6.2.3.2

Verifying that each required RHR pump develops a flow rate ≥ 7700 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by ASME Code, Section XI (Ref. 5). This test confirms one point on the pump performance curve, and the results are indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. UFSAR, Section 14.6.1.3.3.

2. GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, August 1996. | 5

(continued)

BASES

REFERENCES (continued)

3. NEDC-24361-P, James. A FitzPatrick Nuclear Power Plant Suppression Pool Temperature Response, August 1981.
4. 10 CFR 50.36(c)(2)(ii). | J
5. ASME, Boiler and Pressure Vessel Code, Section XI.

Specification 3.6.2.4

A1

3.7 (Cont'd)

4.7 (Cont'd)

[Applicability] (1) The drywell to torus differential pressure shall be established within 24 hours of exceeding 15% rated thermal power during startup. The differential pressure may be reduced to less than the limit up to 24 hours prior to reducing thermal power to less than 15% of rated before a plant shutdown.

[Note to LCO 3.6.2.4] ~~(2)~~ The differential pressure may be decreased to less than 1.7 psid for a maximum of four (4) hours during required operability testing ~~of the HPCI, RCIC, and Suppression Chamber-Drywell Vacuum Breaker System~~. LA1

1 J

[ACTION A] (3) If 3.7.A.7.a above cannot be met, restore the differential pressure to within limits within eight hours

[ACTION B] or reduce thermal power to less than 15% of rated within the next 12 hours.

8. If the specifications of 3.7.A.1 through 3.7.A.5 cannot be met the reactor shall be in the cold condition within 24 hours.

8. Not applicable.

see ITS:
3.6.1.1
3.6.1.2
3.6.1.3
3.6.1.6
3.6.1.7
3.6.2.1
3.6.2.2
and see
as 3.7.A3

Page 2 of 3

Revision J

DISCUSSION OF CHANGES
ITS: 3.6.2.4 - DRYWELL-TO-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted that do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 The reference in CTS 4.7.A.7.a to surveillance requirements of Table 4.2-8 is being deleted since the ITS does not use cross references. The surveillances in current Table 4.2-8 and the proposed Surveillances in ITS 3.3.3.1 are adequate to ensure the instrumentation is functioning properly. Any changes to the current Surveillance Requirements in Table 4.2-8 are discussed in the Discussion of Changes for ITS 3.3.3.1, "Post Accident Monitoring Instrumentation." Since the removal of this cross reference does not change any technical requirements this change is considered administrative and is consistent with the format of NUREG-1433, Revision 1.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS 3.7.A.7.a.(2) allows the differential pressure to be outside its limit for a maximum of 4 hours as a result of required operability testing of HPCI, RCIC, and the Suppresssion Chamber - Drywell Vacuum Breaker System. The details of which Surveillance Tests this allowance is provided for is proposed to be relocated to the Bases. The allowance in the Note to ITS LCO 3.6.2.4 that the limit is not required to be met for 4 hours during Surveillances that cause the drywell-to-suppression chamber differential pressure to be outside the limit is adequate to ensure the allowance is taken only during planned testing. The specific details of the which Operability Note is not necessary to be in the Specification. As such, these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

3.6 CONTAINMENT SYSTEMS

PA1 3.6.2.8 Drywell-to-Suppression Chamber Differential Pressure 4

LCO 3.6.2.8 [CTS 3.7.A.7.a] The drywell pressure shall be maintained ≥ [1.0] psid above the pressure of the suppression chamber. 7 DB1 PA2

Insert Note CLB2

J

APPLICABILITY: MODE 1 during the time period:

[CTS 3.7.A.7.a.1]

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.

CLB1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[CTS 3.7.A.7.a.3] A. Drywell-to-suppression chamber differential pressure not within limit.	A.1 Restore differential pressure to within limit.	8 hours
[CTS 3.7.A.7.a.3] B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ [15]% RTP.	12 hours CLB1

J

[3.7.A.7.a.2]

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
[CTS 4.7.A.7.a] SR 3.6.2.8.1	Verify drywell-to-suppression chamber differential pressure is within limit.	12 hours

4 PA1

Revision J

CLB2

INSERT NOTE

Not required to be met for up to 4 hours during Surveillances that cause or require the drywell-to-suppression chamber differential pressure to be outside the limit.

----

1 3

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.6.2.4 - DRYWELL-TO-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The brackets have been removed and the proper plant specific value has been provided in accordance with CTS 3.7.A.7.a.1 and 3.7.A.7.a.3.

CLB2 The Note to ITS LCO 3.6.2.4 has been added in accordance with CTS 3.7.A.7.a.2 to allow certain required Surveillances to be performed with the limit not met. This allowance is required to perform the test without requiring entry into the Actions. | J

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 ISTS 3.6.2.5 has been renumbered to reflect deletion of ISTS 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray".

PA2 A typographical error has been corrected.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific value has been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.6 Drywell-to-Suppression Chamber Differential Pressure

BASES

BACKGROUND

The toroidal shaped suppression chamber, which contains the suppression pool, is connected to the drywell (part of the primary containment) by [eight] ~~main~~ vent pipes. The ~~main~~ vent pipes exhaust into a continuous vent header, from which [96] downcomer pipes extend into the suppression pool. The pipe exit ~~[3]~~ [4] ft below the minimum suppression pool water level required by LCO 3.6.2.2, "Suppression Pool Water Level." During a loss of coolant accident (LOCA), the increasing drywell pressure will force the waterleg in the downcomer pipes into the suppression pool at substantial velocities as the "blowdown" phase of the event begins. The length of the waterleg has a significant effect on the resultant primary containment pressures and loads.

APPLICABLE SAFETY ANALYSES

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. The required differential pressure results in a downcomer waterleg of ~~[3.06]~~ to ~~[3.58]~~ ft.

Initial drywell-to-suppression chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis ~~Accident~~ LOCA. Drywell-to-suppression chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid.

Drywell-to-suppression chamber differential pressure satisfies Criterion 2 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii) (Ref. 2)

LCO

A drywell-to-suppression chamber differential pressure limit of [1.50] psid is required to ensure that the containment

(continued)

Revision J

BASES

PA4

LCO (continued) 7

conditions assumed in the safety analyses are met. A drywell-to-suppression chamber differential pressure of ≥ [1.60] psid corresponds to a downcomer water leg of [3.58] ft. Failure to maintain the required differential pressure could result in excessive forces on the suppression chamber due to higher water clearing loads from downcomer vents and higher pressure buildup in the drywell.

0.37 ft to 0.49 DB1 5 pipes PA1

Insert LCO if suppression pool level is within the limits specified in LCO 3.6.2.2 PA3 5

CLB2

APPLICABILITY

Drywell-to-suppression chamber differential pressure must be controlled when the primary containment is inert. The primary containment must be inert in MODE 1, since this is the condition with the highest probability for an event that could produce hydrogen. It is also the condition with the highest probability of an event that could impose large loads on the primary containment.

PA1 plant

Inerting primary containment is an operational problem because it prevents primary containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the unit startup and is de-inerted as soon as possible in the unit shutdown. As long as reactor power is < [15]% RTP, the probability of an event that generates hydrogen or excessive loads on primary containment occurring within the first [24] hours following a startup or within the last [24] hours prior to a shutdown is low enough that these "windows," with the primary containment not inerted, are also justified. The [24] hour time period is a reasonable amount time to allow plant personnel to perform inerting or de-inerting.

CLB1

ACTIONS

A.1

If drywell-to-suppression chamber differential pressure is not within the limit, the conditions assumed in the safety analyses are not met and the differential pressure must be restored to within the limit within 8 hours. The 8 hour Completion Time provides sufficient time to restore differential pressure to within limit and takes into account the low probability of an event that would create excessive suppression chamber loads occurring during this time period.

(continued)

Revision 5

CLB2

INSERT LCO

The LCO is modified by a Note which states that the LCO is not required to be met for up to four hours during Surveillances that cause or require drywell-to-suppression chamber differential pressure to be outside of limits. These Surveillances include required OPERABILITY testing of the High Pressure Coolant Injection System, the Reactor Core Isolation Cooling System, and the suppression chamber-to-drywell vacuum breakers. The 4 hour allowance is adequate to perform the Surveillances and to restore the drywell-to-suppression chamber differential pressure to within limits.

PA2

BASES

ACTIONS (continued)

B.1

If the differential pressure cannot be restored to within limits within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by reducing power to ≤ [15]% RTP within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power conditions in an orderly manner and without challenging plant systems. CLB1

SURVEILLANCE REQUIREMENTS

SR 3.6.2.~~6~~ 4.1 PA2

The drywell-to-suppression chamber differential pressure is regularly monitored to ensure that the required limits are satisfied. The 12 hour Frequency of this SR was developed based on operating experience relative to differential pressure variations ~~and pressure instrument drift~~ during applicable MODES ~~and by assessing the proximity to the specified LCO differential pressure limit~~. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal pressure condition. X2 J

REFERENCES

~~None.~~

1. UFSAR, Section 5.2.3.3. DB3 J
2. 10 CFR 50.36 (c)(2)(ii). X1 J

Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.2.4 - DRYWELL-TO-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The brackets have been removed and the proper plant specific value has been provided in accordance with CTS 3.7.A.7.a.1 and 3.7.A.7.a.3.

CLB2 The Note to ITS LCO 3.6.2.4 has been added in accordance with CTS 3.7.A.7.a.2 to allow a certain required Surveillances to be performed with the limit not met. This allowance is required to perform the test without requiring entry into the Actions. The Bases has been revised to reflect this change. J

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA2 ISTS 3.6.2.5 has been renumbered to reflect deletion of ISTS 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray".

PA3 The Bases have been revised for enhanced clarity with no change in intent.

PA4 A typographical error has been corrected.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific value has been provided.

DB2 The brackets have been removed and the word "approximately" included since the value varies depending on suppression pool water level variations.

DB3 The proper Reference has been included.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Drywell-to-Suppression Chamber Differential Pressure

LCO 3.6.2.4 The drywell pressure shall be maintained ≥ 1.7 psi above the pressure of the suppression chamber.

-------------------------NOTE-------------------------
Not required to be met for up to 4 hours during Surveillances that cause or require the drywell-to-suppression chamber differential pressure to be outside the limit.

J

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. Drywell-to-suppression chamber differential pressure not within limit.	A.1	Restore differential pressure to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 15% RTP.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify drywell-to-suppression chamber differential pressure is within limit.	12 hours

J

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.4 Drywell-to-Suppression Chamber Differential Pressure

BASES

BACKGROUND

The toroidal shaped suppression chamber, which contains the suppression pool, is connected to the drywell (part of the primary containment) by eight drywell vent pipes. The drywell vent pipes exhaust into a continuous vent header, from which 96 downcomer pipes extend into the suppression pool. The downcomer pipe exits are approximately 4 ft below the minimum suppression pool water level required by LCO 3.6.2.2, "Suppression Pool Water Level." During a loss of coolant accident (LOCA), the increasing drywell pressure will force the waterleg in the downcomer pipes into the suppression pool at substantial velocities as the "blowdown" phase of the event begins. The length of the waterleg has a significant effect on the resultant primary containment pressures and loads.

APPLICABLE SAFETY ANALYSES

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated downcomer waterleg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown (Ref. 1). The required differential pressure results in a downcomer waterleg of 0.37 ft to 0.49 ft. 3

Initial drywell-to-suppression chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis LOCA. Drywell-to-suppression chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid.

Drywell-to-suppression chamber differential pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

(continued)

BASES (continued)

LCO

A drywell-to-suppression chamber differential pressure limit of 1.7 psi is required to ensure that the containment conditions assumed in the safety analyses are met. A drywell-to-suppression chamber differential pressure of 1.7 psi corresponds to a downcomer water leg of 0.37 ft to 0.49 ft if suppression pool level is within the limits specified in LCO 3.6.2.2. Failure to maintain the required differential pressure could result in excessive forces on the suppression chamber due to higher water clearing loads from downcomer pipes and higher pressure buildup in the drywell.

The LCO is modified by a Note which states that the LCO is not required to be met for up to four hours during Surveillances that cause or require drywell-to-suppression chamber differential pressure to be outside of limits. These Surveillances include required OPERABILITY testing of the High Pressure Coolant Injection System, the Reactor Core Isolation Cooling System, and the suppression chamber-to-drywell vacuum breakers. The 4 hour allowance is adequate to perform the Surveillances and to restore the drywell-to-suppression chamber differential pressure to within limits.

APPLICABILITY

Drywell-to-suppression chamber differential pressure must be controlled when the primary containment is inert. The primary containment must be inert in MODE 1, since this is the condition with the highest probability for an event that could produce hydrogen. It is also the condition with the highest probability of an event that could impose large loads on the primary containment.

Inerting primary containment is an operational problem because it prevents primary containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and is de-inerted as soon as possible in the plant shutdown. As long as reactor power is < 15% RTP, the probability of an event that generates hydrogen or excessive loads on primary containment occurring within the first 24 hours following a startup or within the last 24 hours prior to a shutdown is low enough that these "windows," with the primary containment not inerted, are also justified. The 24 hour time period is a reasonable amount time to allow plant personnel to perform inerting or de-inerting.

(continued)

BASES (continued)

ACTIONS

A.1

If drywell-to-suppression chamber differential pressure is not within the limit, the conditions assumed in the safety analyses are not met and the differential pressure must be restored to within the limit within 8 hours. The 8 hour Completion Time provides sufficient time to restore differential pressure to within limit and takes into account the low probability of an event that would create excessive suppression chamber loads occurring during this time period.

B.1

If the differential pressure cannot be restored to within limits within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by reducing power to ≤ 15% RTP within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.1

The drywell-to-suppression chamber differential pressure is regularly monitored to ensure that the required limits are satisfied. The 12 hour Frequency of this SR was developed based on operating experience relative to differential pressure variations during applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal pressure condition.

REFERENCES

1. UFSAR, Section 5.2.3.3.

2. 10 CFR 50.36(c)(2)(ii).

BASES (continued)

PA1

SURVEILLANCE REQUIREMENTS

SR 3.6.3.~~1~~2.1

Verifying that there is ≥ ~~[4350]~~ 1400 gal of liquid nitrogen supply in ~~the~~ each CAD ~~S~~sub system will ensure at least ~~[7]~~ 3 days of post-LOCA CAD operation. This minimum volume of liquid nitrogen allows sufficient time after an accident to replenish the nitrogen supply for long term inerting. This is verified every 31 days to ensure that the system is capable of performing its intended function when required. The 31 day Frequency is based on operating experience, which has shown 31 days to be an acceptable period to verify the liquid nitrogen supply and on the availability of other hydrogen mitigating systems. DB3

PA1

SR 3.6.3.~~1~~2.2

Verifying the correct alignment for manual, power operated, and automatic valves in each of the CAD subsystem flow paths provides assurance that the proper flow paths exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing.

A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable because the CAD System is manually initiated. This SR 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 position.

The 31 day Frequency is appropriate because the valves are operated under procedural control, improper valve position would only affect a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system.

DB4

REFERENCES

1. ~~Regulatory~~ Safety Guide ~~[2]~~7, ~~Revision [2]~~ March 10, 1971.
2. U FSAR, Section ~~[ ]~~ 5.2.3.8.3. DB5 PA3
3. 10 CFR 50.36 (c)(2)(ii). X1

Revision J

BASES

APPLICABILITY (continued)

if CAD were not available. Therefore, the CAD System is not required to be OPERABLE in MODE 3.

In MODES 4 and 5, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the CAD System is not required to be OPERABLE in MODES 4 and 5.

ACTIONS

A.1

If one CAD subsystem is inoperable, it must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CAD subsystem is adequate to perform the oxygen control function. However, the overall reliability is reduced because a single active failure in the OPERABLE subsystem could result in reduced oxygen control capability. The 30 day Completion Time is based on the low probability of the occurrence of a LOCA that would generate hydrogen and oxygen in amounts capable of exceeding the flammability limit, the amount of time available after the event for operator action to prevent exceeding this limit, and the availability of the OPERABLE CAD subsystem and other hydrogen mitigating systems.

Required Action A.1 has been modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one CAD subsystem is inoperable. This allowance is provided because of the low probability of the occurrence of a LOCA that would generate hydrogen and oxygen in amounts capable of exceeding the flammability limit, the low probability of the failure of the OPERABLE subsystem, the amount of time available after a postulated LOCA for operator action to prevent exceeding the flammability limit, and the availability of other hydrogen mitigating systems.

B.1 and B.2

With two CAD subsystems inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by the Primary Containment Inerting System. The 1 hour Completion Time allows a reasonable period of time to verify that a

(continued)

TA2

INSERT SR 3.6.4.1.4

BWROG-ED-8
~~TSTF-322, Rev.2~~

~~BWR/4 INSERT~~

PA1 The SGT System exhausts the ~~[secondary]~~ containment atmosphere to the environment through appropriate treatment equipment. ~~Each SGT subsystem is designed to draw down pressure in the [secondary] containment to ≥ [0.25] inches vacuum water gauge in ≤ [120] second and maintain pressure in the [secondary] containment at ≥ [0.266] inches vacuum water gauge for 1 hour at a flow rate ≤ [4000] CFM.~~ DB2 To ensure that all fission products released to the ~~[secondary]~~ containment are treated, ~~[SR 3.6.4.1.4 and]~~ SR 3.6.4.1.5 4 verify ies that a pressure in the ~~[secondary]~~ containment that is less than the lowest postulated pressure external to the ~~[secondary]~~ containment boundary can ~~[rapidly]~~ be ~~[established and]~~ maintained. When the SGT System is operating as designed, the ~~establishment and~~ maintenance of ~~[secondary]~~ containment pressure cannot be accomplished if the ~~[secondary]~~ containment boundary is not intact. ~~[Establishment of this pressure is confirmed by SR 3.6.4.1.4 which demonstrates that the [secondary] containment can be drawn down to ≥ [0.25] inches of vacuum water gauge in ≤ [120] seconds using one SGT subsystem.]~~ DB2 SR 3.6.4.1.4 demonstrates that the pressure in the ~~[secondary]~~ containment can be maintained ≥ ~~[0.266]~~ 0.25 inches of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate ≤ ~~[4000]~~ 6000 cfm under calm wind conditions. Calm wind conditions will result in little, if any, infiltration to the secondary containment. Therefore, if the test is performed at other wind conditions and the results are acceptable, this test may be considered met. This test method is acceptable since extreme wind conditions are only expected to be present for a few hours a year. CLB3 ~~The 1 hour test period allows [secondary] containment to be in thermal equilibrium at steady state conditions.~~ The primary purpose of ~~these~~ this SR~~[s]~~ is to ensure ~~[secondary]~~ containment boundary integrity. The secondary purpose of ~~these~~ this SR~~[s]~~ is to ensure that the SGT subsystem being tested functions as designed. ~~There is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the SGT System.~~ DB2 ~~These~~ This SR~~[s]~~ need not be performed for each SGT subsystem. The SGT subsystem used for ~~these~~ this Surveillance~~[s]~~ is staggered to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. The inoperability of the SGT subsystem does not necessarily constitute a failure of ~~these~~ this Surveillance~~[s]~~ relative to the ~~[secondary]~~ containment OPERABILITY. Operating experience has shown the ~~[secondary]~~ containment boundary usually passes ~~these~~ this Surveillance~~[s]~~ when performed at the ~~[18]~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability viewpoint. CLB2

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.2 and SR 3.6.4.1.3 (continued)

outer portion of the barrier closed at all times. However, all secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening.

The 31 day Frequency of SR 3.6.4.1.2 is considered adequate, based on operating experience, and in view of strict administrative procedures required to open a hatch. The 31 day Frequency for SR 3.6.4.1.3 has been shown to be adequate, based on operating experience, and in view of local indication of door status and strict administrative procedures required to be followed for entry and exit.

SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products released to the secondary containment are treated, SR 3.6.4.1.4 verifies that a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary can be maintained. When the SGT System is operating as designed, the maintenance of secondary containment pressure cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.4 demonstrates that the pressure in the secondary containment can be maintained ≥ 0.25 inches of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate ≤ 6000 cfm under calm wind conditions. Calm wind conditions will result in little, if any, infiltration to the secondary containment. Therefore, if the test is performed at other wind conditions and the results are acceptable, this test may be considered met. This test method is acceptable since extreme wind conditions are only expected to be present for a few hours a year. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. The primary purpose of this SR is to ensure secondary containment boundary integrity. The secondary purpose of this SR is to ensure that the SGT subsystem being tested functions as designed. There is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the SGT System. This SR need not be performed for each SGT subsystem. The SGT subsystem used for this Surveillance is staggered to ensure that in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.4 (continued)

addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. The inoperability of the SGT subsystem does not necessarily constitute a failure of this Surveillance relative to the secondary containment OPERABILITY. Operating experience has shown the secondary containment boundary usually passes this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

J

REFERENCES

1. UFSAR, Section 14.6.1.3.

2. UFSAR, Section 14.6.1.4.

3. 10 CFR 50.36(c)(2)(ii).

TECHNICAL CHANGES - MORE RESTRICTIVE

M6 An actual or simulated automatic isolation test (ITS SR 3.6.4.2.3) has been added to the requirements of CTS RETS Table 3.10-2 Item 2 (Refuel Area Exhaust Monitors and Recorders) to ensure both a Logic System Functional Test as well as an actual or simulated automatic isolation test is performed for this Secondary Containment Isolation Instrumentation Function. The new Surveillance will ensure the Function is properly tested throughout their operating sequence. This surveillance is not currently required to be performed, therefore, this change is considered more restrictive on plant operation but is added to enhance plant safety.

M7 ITS SR 3.6.4.2.1, the requirement to verify that each secondary containment isolation manual valve, blind flange, or equivalent that is required to be closed during accident conditions is closed, every 31 days, is being added to CTS 4.7.C. This Surveillance verifies the secondary containment isolation devices are in the correct position to ensure the secondary containment will perform as assumed in the safety analysis. Since the SCIVs are readily accessible to personnel during normal operation and position verification is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions. For clarification Note 1 has been added to the SR which allows the verification of these devices in high radiation areas to be performed by administrative means. This is acceptable since access to these areas is typically restricted during MODES 1, 2 and 3 for ALARA reasons. Note 2 is also included in the SR which does not require the SR to be met for SCIVs that are open under administrative control. This is acceptable since the Bases says that the administrative controls will require stationing a dedicated operator at the controls of the valve who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated. The addition of a new Surveillance Requirement imposes added operational requirements and, therefore, constitutes a more restrictive change. This change is not considered to result in any reduction to safety.

M8 ITS SR 3.6.4.2.2, the requirement to verify that the isolation time of each power operated automatic SCIV is within limits every 92 days, is being added to CTS 4.7.C. This Surveillance verifies the secondary containment isolation valves function to ensure the secondary containment will perform as assumed in the safety analysis. The addition of new Surveillance Requirements imposes additional operational requirements and, therefore, constitutes a more restrictive change. This change is not considered to result in any reduction to safety. 1 J

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
[M7] 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. TA2	31 days
[M8] SR 3.6.4.2.2	Verify the isolation time of each power operated ~~, and each~~ automatic SCIV is within limits. TA1	[In accordance with the Inservice Testing Program or 92 days] X1 J
Table 4.2-1, Table 3.10-2, [L6, L7] SR 3.6.4.2.3	Verify each automatic SCIV actuates to the isolation position on an actual or simulated actuation signal.	$\boxed{18}$ 24 months CLB1

Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.6.4.2 - SECONDARY CONTAINMENT ISOLATION VALVES (SCIVs)

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 ITS SR 3.6.4.2.3 Surveillance Frequency brackets have been removed and the proper value of 24 months included as consistent with CTS RETS Table 3.10-2.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 ITS 3.6.4.2 brackets have been removed and the proper plant specific nomenclature, of Secondary, has been provided with respect to the containment identification.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

None

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 46, Revision 1, have been incorporated into the revised Improved Technical Specifications.

TA2 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 45, Revision 2, have been incorporated into the revised Improved Technical Specifications.

TA3 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 269, Revision 2, have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 ITS SR 3.6.4.2.2 Surveillance Frequency brackets have been removed and the proper value of 92 days included as indicated in M8. J

BASES

APPLICABLE SAFETY ANALYSES (continued)

established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

PA1

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside {secondary} containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (c)(2)(ii) (Ref. 3). X2

LCO

PA1

SCIVs form a part of the {secondary} containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

automatic TA1

The power operated isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference ~~3.~~ 4. X3

I

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference ~~3.~~ 4. X3

APPLICABILITY

PA1

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the {secondary} containment. Therefore, the OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE

(continued)

Revision I

TA2

not locked, sealed, or otherwise secured and is

PA1

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies that each secondary containment manual isolation valve and blind flange that is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the ~~secondary~~ containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in ~~secondary~~ containment that are capable of being mispositioned are in the correct position.

PA1

TA2

INSERT SR 3.6.4.2.1-1

Since these SCIVs are readily accessible to personnel during normal operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions.

Two Notes have been added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these SCIVs, once they have been verified to be in the proper position, is low.

PA5

INSERT SR 3.6.4.2.1-2

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open.

SR 3.6.4.2.2

TA1

Verifying that the isolation time of each power operated, ~~and each~~ automatic SCIV is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that assumed in the safety analyses. The ~~isolation time and~~ Frequency of this SR ~~are~~ ~~in accordance with the Inservice Testing Program or~~ 92 days~~)~~.

PA5

is

X1

J

(continued)

REVISION ~~E~~ J

BASES

SURVEILLANCE REQUIREMENTS (continued) PA1

LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," PA5

SR 3.6.4.2.3 PA1

Verifying that each automatic SCIV closes on a secondary containment isolation signal is required to prevent leakage of radioactive material from ~~[secondary]~~ containment following a DBA or other accidents. This SR ensures that each automatic SCIV will actuate to the isolation position on a ~~[secondary]~~ containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in ~~SR 3.3.6.2.6~~ overlaps this SR to provide complete testing of the safety function. The [18] 24 CLB1 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 24 CLB1

REFERENCES U PA2

1. U FSAR, Section ~~[15.1.39]~~. 14.6.1.3 DB1
2. U FSAR, Section ~~[15.1.41]~~. 14.6.1.4 DB2
3. ~~FSAR, Section [ ].~~ 10 CFR 50.36 (c)(2)(ii). X2
4. Technical Requirements Manual. X3

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Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.4.2 - SECONDARY CONTAINMENT ISOLATION VALVES (SCIVs)

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 ITS SR 3.6.4.2.2 Surveillance Frequency brackets have been removed and the proper value of 92 days included as indicated in M8.

X2 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

X3 ITS 3.6.4.2 has been revised to include reference to the Technical Requirements Manual (TRM).

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
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

1 J

(continued)

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 actuation signal.	24 months

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BASES

APPLICABLE SAFETY ANALYSES (continued)

performs no active function in response to either of these limiting events, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated automatic isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference 4. J

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference 4.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs

(continued)

BASES

ACTIONS

B.1

With two SCIVs in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 4 hours. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 4 hour Completion Time is reasonable considering the time required to isolate the penetration and the probability of a DBA, which requires the SCIVs to close, occurring during this short time, is very low.

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths with two isolation valves. This clarifies that only Condition A is entered if only one SCIV is inoperable in multiple penetrations.

C.1 and C.2

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

D.1, D.2, and D.3

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1 (continued)

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open. These controls consist of stationing a dedicated operator at the controls of the valve who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

SR 3.6.4.2.2

Verifying that the isolation time of each power operated, automatic SCIV is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that assumed in the safety analyses. The Frequency of this SR is 92 days.

SR 3.6.4.2.3

Verifying that each automatic SCIV closes on a secondary containment isolation signal is required to prevent leakage of radioactive material from secondary containment following a DBA or other accidents. This SR ensures that each automatic SCIV will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.7.B.3]	C. (continued)	C.2.1 Suspend movement of irradiated fuel assemblies in {secondary} PA1 containment.	Immediately
		AND	
		C.2.2 Suspend CORE ALTERATIONS.	Immediately
		AND	
[M1]		C.2.3 Initiate action to suspend OPDRVs.	Immediately
[3.7.B.3] [M4] [3.7.B.2.a]	D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately 15
[3.7.B.3] [L1] [M5] [3.7.B.2.b]	E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the {secondary} PA1 containment, during CORE ALTERATIONS, or during OPDRVs.	E.1 ---------NOTE--------- LCO 3.0.3 is not applicable. ---------------------- Suspend movement of irradiated fuel assemblies in {secondary} containment.	Immediately
		AND	
			(continued)

Revision I

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

CLB1

UFSAR Section 16.6

BASES

BACKGROUND

The SGT System is required by ~~10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1)~~. The function of the SGT System is to ensure that radioactive materials that leak from the primary containment into the {secondary} containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. PA1

INSERT BKG-1 DB2

assembly PA4 centrifugal fan PA4

The SGT System consists of two fully redundant subsystems, each with its own set of ductwork, dampers, charcoal filter ~~train~~, and controls.

assembly

SGT Each ~~charcoal~~ filter ~~train~~ consists of (components listed in order of the direction of the air flow): PA4

a. A demister ~~or moisture separator~~;

PA4

b. An electric heater;

c. A prefilter;

d. A high efficiency particulate air (HEPA) filter;

e. A charcoal adsorber;

f. A second HEPA filter; and

g. A centrifugal fan. PA4

DB2

are sized to reduce and maintain

The ~~sizing of the~~ SGT System equipment and components ~~is based on the results of an infiltration analysis, as well as an exfiltration analysis of~~ the {secondary} containment. ~~The internal pressure of the SGT System boundary region is maintained~~ at a negative pressure of {0.25} inches water gauge when the system is in operation, ~~which represents the internal pressure required to ensure zero exfiltration of air from the building when exposed to a [10] mph wind blowing at an angle of [45]° to the building~~. PA1 DB2

DB2

under neutral wind conditions and the SGT fans exhausting at a rate of 6000 cfm

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity

(continued)

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Typ. All Pages

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In addition, the OPERABILITY of each SGT decay heat cooling valve is verified to ensure the valve closes on subsystem initiation (interlocked with the suction valve) and opens when shutdown. This will ensure the mitigation functions as well as the decay heat cooling mode of each SGT subsystem is available.

DB7

1J

SGT System
B 3.6.4.3

1J

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.3

This SR verifies that each SGT subsystem starts on receipt of an actual or simulated initiation signal. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the [18] 24 month Frequency. The LOGIC SYSTEM FUNCTIONAL TEST in ~~SR 3.3.6.2.6~~ LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

PA2 CLB2

SR 3.6.4.3.4

This SR verifies that the filter ~~cooler bypass damper can be opened and the fan started~~ cooling cross-tie valves are OPERABLE. This ensures that the ~~ventilation~~ decay heat cooling mode of SGT System operation is available. ~~While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency, which is based on the refueling cycle. Therefore, the Frequency was found to be acceptable from a reliability standpoint.~~

INSERT SR 3.6.4.3.4

CLB3 CLB1 CLB3

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 41~~ UFSAR, Section 16.6

2. UFSAR, Section ~~[6.2.3]~~ 5.3.3.4

3. ~~Regulatory Guide 1.52, Rev. [2]~~ UFSAR, Section 14.6

4. 10 CFR 50.36 (c)(2)(ii).

DB5 PA4 TA1 DB6 X1

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend movement of irradiated fuel assemblies in 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 secondary containment. AND	Immediately
	E.2 Suspend CORE ALTERATIONS. AND	Immediately
	E.3 Initiate action to suspend OPDRVs.	Immediately

BASES

BACKGROUND (continued)

d. A high efficiency particulate air (HEPA) filter;

e. A charcoal adsorber; and

f. A second HEPA filter.

The SGT System equipment and components are sized to reduce and maintain the secondary containment at a negative pressure of 0.25 inches water gauge when the system is in operation under neutral wind conditions and the SGT fans exhausting at a rate of 6,000 cfm. 1

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, both SGT subsystem fans start. Upon verification that both subsystems are operating, one subsystem is normally shut down.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and refueling accidents (Ref. 3). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1 (continued)

vibration can be detected for corrective action. Operation with the heaters on for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

This SR verifies that each SGT subsystem starts on receipt of an actual or simulated initiation signal. In addition, the OPERABILITY of each SGT decay heat cooling valve is verified to ensure the valve closes on subsystem initiation (interlocked with the suction valve) and opens when shutdown. This will ensure the mitigation function as well as the decay heat cooling mode of each SGT subsystem is available. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.4

This SR verifies that the filter cooling cross-tie valves are OPERABLE. This ensures that the decay heat cooling mode of SGT System operation is available. The 24 month Frequency has been shown to be adequate, based on operating experience, and in view of the strict administrative controls required for entry into the area of these valves. 5

REFERENCES

1. UFSAR, Section 16.6.
2. UFSAR, Section 5.3.3.4.
3. UFSAR, Section 14.6.
4. 10 CFR 50.36(c)(2)(ii).

SUMMARY OF CHANGES TO ITS SECTION 3.7 - REVISION J

Source of Change	Summary of Change	Affected Pages
Retyped ITS typographical errors	Minor typographical errors in the retyped ITS have been corrected to be consistent with the NUREG markup. (The title of LCO 3.4.7 has been deleted from the Note to ITS 3.7.1 Required Action D.1; and the word "Core" has been changed to "CORE" in the ITS 3.7.4 Applicability.)	**Specification 3.7.1** Retyped ITS p 3.7-2 **Specification 3.7.4** Retyped ITS p 3.7-11
NUREG ITS markup error	A minor NUREG markup error has been corrected to be consistent with the retyped ITS. (A period has been added to the end of the first Condition of INSERT ACTION B.)	**Specification 3.7.2** NUREG ITS markup p Insert page 3.7-4
Retyped ITS Bases typographical errors	Minor typographical errors in the retyped ITS Bases have been corrected to be consistent with the NUREG Bases markup. (The word "active" has been added to the ITS 3.7.1 ASA section; the word "lake" has been added to the ITS 3.7.2 Background section; one paragraph has been split into two paragraphs and the words "based on engineering judgment, is" have been added to the SR 3.7.2.4 section; the words "Control Room" have been decapitalized (three places) in the ITS 3.7.4 Background section; the word "Water" has been added between the words "Service" and "System" in the ITS 3.7.4 LCO section; the word "recombiner" has been changed to "SJAE" in the ITS 3.7.5 Background section; the word "approximately" has been added to the SR 3.7.5.1 section; the word "analysis" has been changed to "analyses" in the ITS 3.7.6 Applicability section; the words "applicable safety analyses" have been changed to "abnormal operational" in the ITS 3.7.6 Actions B.1 section; and the word "meets" has been changed to "ensures that" and the words "are met" have been added in the ITS 3.7.7 Background section.)	**Specification 3.7.1** Retyped ITS Bases p B 3.7-2 **Specification 3.7.2** Retyped ITS Bases p B 3.7-7 and B 3.7-13 **Specification 3.7.4** Retyped ITS Bases p B 3.7-24 **Specification 3.7.5** Retyped ITS Bases p B 3.7-29 and B 3.7-31 **Specification 3.7.6** Retyped ITS Bases p B 3.7-34 and B 3.7-35 **Specification 3.7.7** Retyped ITS Bases p B 3.7-37

SUMMARY OF CHANGES TO ITS SECTION 3.7 - REVISION J

Source of Change	Summary of Change	Affected Pages
NUREG Bases markup errors	Minor NUREG Bases markup errors have been corrected to be consistent with the retyped ITS Bases. (Periods have been added to the ITS 3.7.1 References 4 and 6; the word "theis" has been changed to "this" in the ITS 3.7.2 Actions A.1 section; the word "surveillance" has been capitalized" in the SR 3.7.2.7 section; a period has been added to ITS 3.7.2 Reference 4; the words "(Ref. 6)" have been changed to "(Ref. 5)" in the ITS 3.7.3 ASA section; periods have been added to ITS 3.7.3 References 3 and 5; the words "Control Room" have been decapitalized in ITS 3.7.4 INSERT LCO; a period has been added to the SR 3.7.4.1 section; a period has been added to ITS 3.7.4 Reference 2; a period has been added to the SR 3.7.5.1 section; periods have been added to ITS 3.7.5 References 3 and 4; and a comma has been added to ITS 3.7.7 Reference 4.)	**Specification 3.7.1** NUREG Bases markup p B 3.7-6 **Specification 3.7.2** NUREG Bases markup p B 3.7-10 and B 3.7-13 **Specification 3.7.3** NUREG Bases markup p B 3.7-19 and B 3.7-24 **Specification 3.7.4** NUREG Bases markup p Insert Page B 3.7-26 and B 3.7-29 **Specification 3.7.5** NUREG Bases markup p B 3.7-32 **Specification 3.7.7** NUREG Bases markup p B 3.7-39
Typographical errors	Minor typographical errors have been corrected in the NUREG ITS markup and the retyped ITS. (A comma has been deleted from the ITS 3.7.4 Background Bases section; and periods have been added to Condition A and Required Action A.1.)	**Specification 3.7.4** NUREG Bases markup p B 3.7-25 Retyped ITS Bases p B 3.7-23 **Specification 3.7.6** NUREG ITS markup p 3.7-18 Retyped ITS p 3.7-16

SUMMARY OF CHANGES TO ITS SECTION 3.7 - REVISION J

Source of Change	Summary of Change	Affected Pages
Consistency issues	Minor consistency issue corrections have been made. (The word "required" has been added to ITS 3.7.2 Required Action B.1 for consistency with the usage throughout the ITS, since not all deicing heaters are required to be Operable; since a single division of deicing heaters will never result in a single EDG subsystem being inoperable (both divisions of deicing heaters are on a common suction header), the Note to ITS 3.7.2 Required Action B.1 is not necessary and has been deleted; a reference to LCO 3.7.4 has been added to the ITS 3.7.2 Applicability Bases, since ESW supports the Control Room AC System Operability; the LCO 3.3.7.3 title has been deleted from the SR 3.7.2.7 Bases since it is listed earlier in the Bases; the word "main" in the LCO Note added by TSTF-287 (and in the corresponding Bases), used when referring to the control room (i.e., "main" control room), has been deleted to be consistent with plant nomenclature; the term "emergency booster fan" in ITS 3.7.3 Bases (two places) has been changed to "control room emergency air supply fan" to be consistent with plant terminology; the words ", as indicated by the SJAE monitor," (which are consistent with the CTS) have been added to the SR 3.7.5.1 Bases to describe where to determine the gross gamma activity rate; the word "The" in the title of ITS 3.7.6 has been deleted.)	**Specification 3.7.2** NUREG ITS markup p Insert Page 3.7-4 NUREG Bases markup p Insert Page B 3.7-9, Insert Page B 3.7-10, and B 3.7-13 Retyped ITS p 3.7-3 Retyped Bases p B 3.7-10, B 3.7-11, and B 3.7-13 **Specification 3.7.3** NUREG ITS markup p 3.7-9 NUREG Bases markup p B 3.7-18, B 3.7-19, Insert Page B 3.7-19, and Insert Page B 3.7-20 Retyped ITS p 3.7-7 Retyped Bases p B 3.7-15, B 3.7-16, B 3.7-17, and B 3.7-18 **Specification 3.7.5** NUREG Bases markup p B 3.7-32 Retyped Bases p B 3.7-31 **Specification 3.7.6** NUREG ITS markup p 3.7-18 Retyped ITS p 3.7-16

SUMMARY OF CHANGES TO ITS SECTION 3.7 - REVISION J

Source of Change	Summary of Change	Affected Pages
Consistency issue	The Main Turbine Bypass System includes four main turbine bypass valves, but only three of the four valves are required for the System to be considered Operable. This is stated in the LCO section of the Bases. For consistency, the word "required" has been added to SR 3.7.5.1, which requires the main turbines bypass valves to be cycled, since only three of the four are required Operable. The ACTION A.1 Bases has also been modified to state that the Main Turbine Bypass System is inoperable when two or more bypass valves are inoperable. Also, the word "assumed" has been added to the Turbine Bypass System Response Time definition, since the turbine bypass capacity referenced in the definition is from only three of the valves, not all four.	**Specification 3.7.6** NUREG markup p 3.7-18 JFD DB2 (JFDs p 1 of 2) NUREG Bases markup p B 3.7-34 and B 3.7-35 Retyped ITS p 3.7-17 Retyped Bases p B 3.7-34 and B 3.7-35 **Specification 1.1** NUREG markup p 1.1-7 JFD X3 (JFDs p 5 of 5) Retyped ITS p 1.1-6
Editorial change	It was noted that RETS 3.5.b was deleted via an "R" DOC in ITS 3.7.5. However, in the Split Report, this was not described. Therefore, a proper discussion of why the SJAE Radiation Monitors do not meet the criteria of 10 CFR 50.36 has been added to the Split Report. The new information in the Split Report is derived from the "R" DOC in ITS 3.7.5.	**Split Report** Summary Disposition Matrix p 12 of 14 Appendix A p 23a of 23
Editorial change	A clarification has been added to the ITS 3.7.2 LCO Bases, which describes that Operability of the ESW pumps (with respect to flow rates) is based on measured performance during IST testing. This is consistent with current plant practice. Currently, the NUREG does not specify any flow rate requirements for the pumps.	**Specification 3.7.2** NUREG Bases markup p B 3.7-8 and Insert Page B 3.7-8 Retyped ITS Bases p B 3.7-9
Editorial change	The Note to ISTS SR 3.7.2.4 states that isolation of flow to individual components does not render [PSW] System (ESW System in the JAFNPP ITS) inoperable. This Note was added to allow the individual components whose cooling water has been isolated to be declared inoperable in lieu of declaring the ESW System inoperable. However, it was added as an allowance, such that the ESW System could be declared inoperable when individual components are isolated and the ACTIONS of LCO 3.7.2 taken. For example, if all the components cooled by ESW have their cooling water isolated, the proper action would be to declare the ESW System inoperable. For clarity, the word "necessarily" has been added to the Note to ensure that it is always an option to declare the ESW System inoperable.	**Specification 3.7.2** DOC A7 (DOCs p 2 of 9) NUREG ITS markup p 3.7-6 JFD PA3 (JFDs p 1 of 2) NUREG Bases markup p B 3.7-12 Bases JFD PA1 (Bases JFDs p 1 of 3) Retyped ITS p 3.7-5 Retyped ITS Bases p B 3.7-13

SUMMARY OF CHANGES TO ITS SECTION 3.7 - REVISION J

Source of Change	Summary of Change	Affected Pages
Editorial change	The Background, LCO, ACTIONS, and SR section of ITS 3.7.2 Bases has been modified to more clearly reflect the ESW design basis.	**Specification 3.7.2** NUREG Bases markup p Insert Page B 3.7-7a, Insert Page B 3.7-7b, Insert Page B 3.7-8, Insert Page B 3.7-10, and Insert Page B 3.7-12 Retyped ITS Bases p B 3.7-8, B 3.7-9, B 3.7-10, B 3.7-11, and B 3.7-12
Editorial change	The ITS 3.7.3 Bases Background section has been modified to more clearly reflect the CREVAS System design basis.	**Specification 3.7.3** NUREG Bases markup p B 3.7-18 Retyped ITS Bases p B 3.7-16
Editorial change	The Reference for the Supplemental Reload Licensing Report has been modified to state that the current revision number is located in the COLR, in lieu of listing the current revision number in the ITS Bases. This will preclude requiring a Bases change after every refueling outage.	**Specification 3.7.6** NUREG Bases markup p Insert Page B 3.7-36 Retyped ITS Bases p B 3.7-36
Technical change	The maximum suppression pool water temperture during the assumed accident has been changed from 209 degrees F to 213 degrees F, based on the most recently applicable calculations in the UFSAR. In addition, a Reference that is not necessary anymore has been deleted and the remaining References renumbered as necessary.	**Specification 3.7.1** NUREG Bases markup p B 3.7-2 and B 3.7-6 Retyped ITS Bases p B 3.7-2 and B 3.7-6
Technical change	The CREVAS System flow rate requirement in SR 3.7.3.3 has been changed from $\leq$ 1100 scfm to $\geq$ 900 scfm and $\leq$ 1100 scfm, consistent with the current licensing basis in CTS 4.11.A.5. Appropriate DOC, NSHC, and Bases changes are also made.	**Specification 3.7.3** CTS markup p 3 of 3 DOCs M5 and L1 (DOCs p 4 of 8 and 8 of 8) NSHC L1 (NSHCs p 1 of 2 and 2 of 2 NUREG ITS markup p 3.7-12 NUREG Bases markup p B 3.7-24 Retyped ITS p 3.7-10 Retyped ITS Bases b B 3.7-22

SUMMARY OF CHANGES TO ITS SECTION 3.7 - REVISION J

Source of Change	Summary of Change	Affected Pages
Technical change	CTS 4.11.A.4 requires a 24 month calibration of CREVAS System temperature transmitters and differential pressure switches. DOC LB1 justifies the relocation of CTS 4.11.A.4 to the TRM, since the instruments are related to the AC portion of the CREVAS System, and are not necessarily required to ensure proper operation of the Control Room AC System. The LB1 DOC has been modified to delete the CR Exhaust Fan inlet temperature switch (which was the last instrument in the Table presented in the DOC), since the CR Exhaust Fan inlet temperature switch is not required to be tested by CTS 4.11.A.4 (it is neither a temperature transmitter nor a differential pressure switch).	**<u>Specification 3.7.3</u>** DOC LB1 (DOCs p 6 of 8)
Technical change	CTS RETS 3.5.a (LCO portion) states that the gross radioactivity rate of the noble gases is measured at the discharge of the SJAE. However, the Surveillance requirement of CTS RETs 3.5.a provides two locations to measure the gross radioactivity rate of the noble gases; at the discharge of the SJAE (prior to dilution and/or discharge) or at the recombiner discharge (prior to the delay of the offgas to reduce the total radioactivity). In the original submittal, the ITS LCO 3.7.5 was modified to list both locations. However, currently the only sample location used by JAFNPP to meet the CTS RETS requirement is the discharge of the SJAE. Therefore, the second method has been deleted from the ITS LCO, and the limit of the LCO will always be met by sampling the discharge of the SJAE.	**<u>Specification 3.7.5</u>** CTS p 1 of 6 DOCs M2 and LA1 (DOCs p 1 of 4 and 2 of 4) NUREG ITS markup p Insert Page 3.7-16 NUREG Bases markup p B 3.7-32 Retyped ITS p 3.7-14 Retyped ITS Bases p B 3.7-31

BASES (continued)

Via the RHR System (PA2)

APPLICABLE SAFETY ANALYSES

The RHRSW System removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its function of limiting the release of radioactive materials to the environment following a LOCA. The ability of the RHRSW System to support long term cooling of the reactor or primary containment is discussed in the FSAR, ~~Chapters~~ [6] and [15] (Refs. 2 and 8, respectively). These analyses explicitly assume that the RHRSW System will provide adequate cooling support to the equipment required for safe shutdown. These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

PA1 (1) Sections 4.8, 5.1 (14) Chapter DB4 (3) (4)

The safety analyses for long term cooling were performed for various combinations of RHR System failures. The worst case single active failure that would affect the performance of the RHRSW System is any failure that would disable one subsystem of the RHRSW System. As discussed in the FSAR, Section [6.2.1.4.3] 14.6.1.3.3 (Ref. 6) for these analyses, manual initiation of the OPERABLE RHRSW subsystem and the associated RHR System is assumed to occur [10] minutes after a DBA. The RHRSW flow assumed in the analyses is [4000] gpm per pump with two pumps operating in one loop. In this case, the maximum suppression chamber water temperature ~~and pressure are~~ [206.4]°F 213°F ~~and [36.59] psig, respectively~~, which is ~~well~~ below the design temperature of [340] 220 °F ~~and maximum allowable pressure of [62] psig~~.

PA2 active PA1 (1) (5) DB4 DB1 DB4 DB1 (15) DB5 PA2 DB1

The RHRSW System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 6).

X1 J J

LCO

Two RHRSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power.

An RHRSW subsystem is considered OPERABLE when:

a. Two pumps are OPERABLE; and

(continued)

Revision J

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

REFERENCES PA1 DB4

1. U FSAR, Section ~~[9.2.7]~~ 7.3.
2. FSAR, ~~Chapter [6]~~ Section 4.8.
3. FSAR, ~~Chapter [15]~~ Section 5.1. DB4 J
4. UFSAR, Chapter 14. DB4
5. U FSAR, Section ~~[6.2.1/4.3]~~ 14.6.1.3.3. PA1
6. 10 CFR 50.36(c)(2)(ii). X1 J

Revision J

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Both RHRSW subsystems inoperable for reasons other than Condition B.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.4.7 for RHR shutdown cooling made inoperable by RHRSW System. ----------------------------	
	D.1 Restore one RHRSW subsystem to OPERABLE status.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	E.2 Be in MODE 4.	36 hours

J

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each RHRSW 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

BASES

APPLICABLE SAFETY ANALYSES (continued)

System to support long term cooling of the reactor or primary containment is discussed in the UFSAR, Sections 4.8, 5.1 and Chapter 14 (Refs. 2, 3 and 4, respectively). These analyses explicitly assume that the RHRSW System will provide adequate cooling support to the equipment required for safe shutdown. These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The safety analyses for long term cooling were performed for various combinations of RHR System failures. The worst case single active failure that would affect the performance of the RHRSW System is any failure that would disable one subsystem of the RHRSW System. As discussed in the UFSAR, Section 14.6.1.3.3 (Ref. 5) for these analyses, manual initiation of the OPERABLE RHRSW subsystem and the associated RHR System is assumed to occur 10 minutes after a DBA. The RHRSW flow assumed in the analyses is 4000 gpm per pump with two pumps operating in one loop. In this case, the maximum suppression chamber water temperature is 213°F, which is below the design temperature of 220°F.

The RHRSW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 6).

LCO

Two RHRSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power.

An RHRSW subsystem is considered OPERABLE when:

a. Two pumps are OPERABLE; and

b. An OPERABLE flow path is capable of taking suction from the intake structure and transferring the water to the RHR heat exchangers at the assumed flow rate and discharging the water to the discharge structure.

The requirements of the ultimate heat sink are not addressed in this LCO since the requirements of the ultimate heat sink are addressed by the emergency service water pump

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

REFERENCES

1. UFSAR, Section 9.7.3.
2. UFSAR, Section 4.8.
3. UFSAR, Section 5.1.
4. UFSAR, Chapter 14.
5. UFSAR, Section 14.6.1.3.3.
6. 10 CFR 50.36(c)(2)(ii).

DISCUSSION OF CHANGES
ITS: 3.7.2 - EMERGENCY SERVICE WATER (ESW) SYSTEM AND
ULTIMATE HEAT SINK (UHS)

ADMINISTRATIVE CHANGES

A4 (continued)

The deicing heaters are considered to be part of the UHS. This change is consistent with the format of NUREG-1433, Revision 1.

A5 CTS 3.11.E requires the intake deicing heaters to be Operable when intake water temperature is less than or equal to 37°F. When these heaters are inoperable the default action is to be in cold conditions (CTS 3.11.E). In ITS 3.7.1, the Applicability of the deicing heaters is MODE 1, 2 and 3 consistent with the requirements of the Emergency Service Water (ESW) System. A Note has been added to the applicable surveillances related to the heaters (SR 3.7.2.3, SR 3.7.2.5 and SR 3.7.2.6) that these SRs are not required to be met at lake temperatures > 37°F. Since the Applicability of when the heaters are required to be Operable is consistent with the CTS, this change is considered administrative.

A6 CTS 4.11.E.1 requires the weekly verification of the six heater feeder ammeters. ITS SR 3.7.2.3 requires the verification of the "required" deicing heater feeder current for each division of deicing heaters. Since CTS 3.11.E only requires 18 out of 88 heaters to be OPERABLE, there is no reason to require the measurement of all heater feeder ammeters (6 per design) since the CTS LCO can be met with only one set of heaters (Division 1 or 2) in operation. A description of the method to satisfy the requirement is included in the Bases for SR 3.7.2.5. In addition, the word "required" has been added to CTS 4.11.E.1, 4.11.E.2 and 4.11.E.3 (SR 3.7.2.3, SR 3.7.2.5 and SR 3.7.2.6, respectively). Since this change simply provides consistency between the requirements in the LCO (CTS 3.11.E) and the CTS Surveillance, this change is considered administrative. In addition, a more restrictive change (M3) adds the requirement that both divisions of deicing heaters are required.

A7 A Note (Note to ITS SR 3.7.2.4) has been added to CTS 4.11.D.1.c (the valve alignment verification Surveillance) which clarifies that the isolation of flow to individual components does not necessarily render ESW System inoperable. The isolation of individual components does not necessarily place the ESW System in an inoperable state. The ESW System may still be capable of providing cooling water to OPERABLE safety related components, however the OPERABILITY of these individual components which have been isolated must be considered. The OPERABILITY of each individual component of the ESW will be accounted for within the OPERABILITY requirements of the associated supported system Specification within the ITS. This is consistent with current practice and is based on the definition of Operable in CTS definition 1.0.J in

ADMINISTRATIVE CHANGES

A7 (continued)

the ITS definition of OPERABLE - OPERABILITY in ITS Section 1.0 which require cooling water to be available for a system, subsystem, division, component, or device to be considered OPERABLE to perform its specified safety function. Since this Note is only added for clarity, this change is considered administrative. This change is consistent with NUREG-1433, Revision 1.

A8 The requirements in CTS 4.11.E.2 to monitor the individual heater current once every 6 months has been changed to require the verification of the required deicing heater power (ITS SR 3.7.2.5). The current is measured more frequently in CTS 4.11.E.1. This Surveillance ensures that the required deicing heaters are operating as designed ensuring the appropriate power is produced in each required heater. Since this change is consistent with current practice, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.11.D.3 requires the reactor to be placed in a cold condition within 24 hours if the requirements of CTS 3.11.D.2 (one ESW subsystem inoperable) can not be met. CTS 3.11.E.1 requires the same actions when the required deicing heaters are found to be inoperable (see M3 for inclusion of redundant deicing heater divisions). CTS 3.11.D.1 requires both ESW subsystems to be Operable, except as allowed by CTS 3.11.D.2. CTS 3.11.D.2 addresses the condition with one inoperable ESW subsystem. Therefore, With two inoperable ESW subsystems entry into CTS 3.0.C is required and the plant must be in COLD SHUTDOWN within 24 hours. In ITS 3.7.2, all default actions for the ESW System and ultimate heat sink (UHS) are covered in ACTION C for clarity consistent with the format of NUREG-1433, Revision 1. An additional ACTION has been added to allow time to restore a division of inoperable deicing heaters to Operable status (ACTION B), however this change is addressed in M3. The inoperability of two ESW subsystems is addressed in the second part of Condition B. If the Required Action and associated Completion Time of ACTION A (for one ESW subsystem) or ACTION B (for one division of deicing heaters) is not met entry into the first part of Condition B is required. Finally, if the ultimate heat sink (UHS) is inoperable for reasons other than one division of deicing heaters, entry into the third part to Condition C is required. However this requirement was added in accordance with M2.

DB3

INSERT ACTION B

B.	One division of required deicing heaters inoperable. AND UHS temperature ≤ 37°F.	B.1	Restore the division of required deicing heaters to OPERABLE status.	7 days

J

E DB1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.7.2.4~~	~~Operate each [PSW] cooling tower fan for ≥ [15] minutes.~~	~~31 days~~
SR 3.7.2.5 4	necessarily PA3 ------------------NOTE------------------ Isolation of flow to individual components does not render PSW System inoperable. --- E Verify each PSW subsystem manual, power operated, and automatic valve in the flow paths servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days DB1
SR 3.7.2.6 7	E DB1 Verify each PSW subsystem actuates on an actual or simulated initiation signal.	[18] months 24 CLB2

DB3 CLB3 DB3 [4.11.D.1.c] DB3 CLB3 [4.11.D.1.a] 5

Insert SR-2 CLB3 Insert SR-1

Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.7.2 - EMERGENCY SERVICE WATER (ESW) SYSTEM AND
ULTIMATE HEAT SINK (UHS)

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The ITS 3.7.2 ACTIONS A Completion Time of 7 days is consistent with the current licensing basis (CTS 3.11.D.2) and with the Completion Time of an inoperable emergency diesel generator subsystem in ITS 3.8.1.

CLB2 The brackets have been removed the proper value included. The ITS SR 3.7.2.7 Frequency of 24 months is consistent with the current licensing basis (CTS 4.11.D.1.a).

CLB3 Three additional Surveillance Requirements have been added to ITS 3.7.2 consistent with the existing requirements in CTS 4.11.E. Subsequent SRs have been renumbered as required.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Editorial changes have been made to be consistent with the Writers Guide.

PA2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA3 The Note to ISTS SR 3.7.2.4 states that isolation of flow to individual components does not render [PSW] System (ESW System in the JAFNPP ITS) inoperable. This Note was added to allow the individual components whose cooling water has been isolated to be declared inoperable in lieu of declaring the ESW System inoperable. However, it was added as an allowance, such that the ESW System could be declared inoperable when individual components are isolated and the ACTIONS of LCO 3.7.2 taken. For example, if all the components cooled by ESW have their cooling water isolated, the proper action would be to declare the ESW System inoperable. For clarity, the word "necessarily" has been added to the Note to ensure that it is always an option to declare the ESW System inoperable. J

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific information/value has been provided.

DB2 ISTS 3.7.2 ACTIONS A and B have been deleted since each ESW subsystem at JAFNPP has only one pump. Subsequent ACTIONS have been renumbered and modified, as applicable.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.7.2 - EMERGENCY SERVICE WATER (ESW) SYSTEM AND
ULTIMATE HEAT SINK (UHS)

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB3 ISTS 3.7.2 ACTION C and ISTS SRs 3.7.2.1 and 3.7.2.4 are being deleted because the design of JAFNPP Emergency Service Water System does not include cooling towers. However, ACTION B has been added to cover the condition where one division of required deicing heaters is inoperable. Subsequent ACTIONS and SRs have also been renumbered and modified, as applicable.

DB4 ISTS 3.7.2 Required Action D.1 Note 2 has been deleted since an inoperable ESW subsystem does not necessarily make RHR Shutdown Cooling System inoperable. ESW provides cooling to the crescent area coolers which supports the Operability of the RHR pumps, however the cooling capacity of the other crescent area coolers can provide support to all RHR pumps. The Safety Function Determination Program will be implemented at ITS implementation as required by LCO 3.0.6 and described in Specification 5.5.12. This program will provide the appropriate guidance for entry into the applicable Conditions and Required Actions of LCO 3.4.7 upon loss of the cooling function, therefore the deletion of this Note is considered acceptable.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

INSERT BKGRD 1

Each ESW pump will automatically pump to the associated EDG cooler. The remaining ESW loads will be automatically cooled when the associated ESW supply header isolation valve opens and the associated ESW minimum flow valve closes. This occurs when the ESW lockout matrix logic actuates upon low reactor building closed loop cooling water pump discharge pressure. This logic is discussed in LCO 3.3.7.3, "Emergency Service Water (ESW) System Instrumentation". In addition, the ESW pumps will automatically start in response to the ESW lockout matrix logic. However, this function is not required for safe reactor shutdown since the ESW pumps will start when any associated EDG starts.

INSERT BKGRD 2

The ESW System is described in UFSAR, Section 9.7.1 (Ref. 1).

INSERT BKGRD 3

The lake intake structure is a reinforced concrete structure sitting on the lake bottom at a distance of approximately 900 ft from the shoreline in approximately 25 ft of water. The top surface of the intake structure is at the 233 ft elevation (above sea level), which is approximately 10 ft below the historically lowest monthly mean lake level. The intake is a roofed structure which draws water in through side openings that are protected with bar racks spaced at 1 ft centers to block the entrance of large debris. This results in water being taken in at lower levels and prevents the formation of vortices at the surface, thus minimizing the possibility of floating ice being drawn down from the surface. The side intake area of approximately 8 ft by 70 ft, less the bar rack area, provides a net clear area of 552 ft^2. During normal operation, with a maximum nominal operating flow of 388,600 gpm from three circulating water pumps and two normal service water pumps, the average intake velocity is approximately 1.6 ft per second. However, during safe shutdown conditions with only two Residual Heat Removal Service Water (RHRSW) pumps and one ESW pump in operation, the maximum nominal flow is reduced to 10,000 gpm, corresponding to an average intake velocity of 0.04 ft per second.

The formation of frazil ice on the steel bar racks at the intake structure openings is common in northern climates. This kind of ice is formed when meteorological conditions are such that the water is subcooled below its freezing point due to radiational cooling. Under these conditions, frazil ice can form on intake bar racks or spongy masses of this ice, formed in other parts of the lake and carried past an intake by wind-driven currents, can adhere to the bar racks. Sufficient transport velocity exists to move buoyant frazil ice from the lake surface to the intake structure during normal operation, but not under safe shutdown conditions. If ice formation does occur on the bar racks during normal operation, sufficient local erosion velocities will develop to limit total ice accumulation such that the remaining net clear intake area would be sufficient to meet required safe shutdown flows. In an effort to suppress the formation of frazil ice on the bar

<u>INSERT BKGD 3</u> (continued)

racks, each of the 88 rack bars is heated by a deicing heater. Each deicing heater is rated at 1670 watts and is normally energized. Forty four heaters are powered by one division while the remaining 44 heaters are powered by the other division.

BASES

APPLICABLE SAFETY ANALYSES (continued)

analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The ability of the [PSW] System to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analyses evaluated in References 2 and 3. The ability to provide onsite emergency AC power is dependent on the ability of the [PSW] System to cool the DGs. ~~The long term cooling capability of the RHR, core spray, and RHR service water pumps is also dependent on the cooling provided by the [PSW] System.~~ Insert ASA

The [PSW] System, together with the [UHS], satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The [PSW] subsystems are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one subsystem of [PSW] is required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, two subsystems of [PSW] must be OPERABLE. At least one subsystem will operate, if the worst single active failure occurs coincident with the loss of offsite power.

A subsystem is considered OPERABLE when it has an OPERABLE one [UHS], ~~two~~ OPERABLE pumps, and an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the appropriate equipment. Insert LCO-1

The OPERABILITY of the [UHS] is based on having a minimum water level in the ~~pump well of the~~ screenwell ~~intake structure~~ of ~~[60.7]~~ 236.5 ft mean sea level and a maximum water temperature of ~~[90]~~ 85 °F.

Insert LCO

The isolation of the [PSW] System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the [PSW] System.

APPLICABILITY

In MODES 1, 2, and 3, the [PSW] System and [UHS] are required to be OPERABLE to support OPERABILITY of the

(continued)

Revision I

DBI

Insert ASA

The long term cooling capability of RHR and core spray pumps is dependent on the capability of the ESW System to provide cooling to the EDGs as well as the crescent area coolers.

DB12

INSERT LCO

With UHS temperature ≤ 37°F, conditions become increasingly favorable for the formation of frazil ice on the intake structure bar racks during normal operation. Therefore, in an effort to suppress the formation of frazil ice on the intake structure bar racks, at least 18 out of the 44 deicing heaters (each heater producing 1670 watts) in each electrical division are maintained OPERABLE whenever UHS temperature is ≤ 37°F. J

PA3

INSERT LCO-1

OPERABILITY of equipment cooled by the ESW System is based on heat transfer, not flow rates; OPERABILITY of the ESW pumps is based on measured performance remaining within allowable IST Program acceptance criteria. J

PA3 INSERT APP

and therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. Thus, LCO 3.7.4, "Control Room AC System," and LCO 3.8.2, "AC Sources - Shutdown," which require the ESW System to be OPERABLE, will govern ESW System operation in MODES 4 and 5. J

BASES

ACTIONS

~~C.1 (continued)~~

~~the time required to reasonably complete the Required Action.~~

B.1

With one [PSW] subsystem inoperable ~~for reasons other than Condition A and [Condition C] (e.g., inoperable flow path or both pumps inoperable in a loop)~~, the [PSW] subsystem must be restored to OPERABLE status within ~~72 hours~~ 7 days. With the ~~unit~~ plant in this condition, the remaining OPERABLE [PSW] subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single active component failure in the OPERABLE [PSW] subsystem could result in loss of [PSW] function.

The ~~72 hour~~ 7 day Completion Time is based on the redundant [PSW] System capabilities afforded by the OPERABLE subsystem, the low probability of an accident occurring during this time period, and is consistent with the allowed Completion Time for restoring an inoperable DG subsystem.

Required Action B.1 is modified by ~~two~~ a Notes indicating that the applicable Conditions of LCO 3.8.1, "AC Sources—Operating," ~~LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown,"~~ be entered and Required Actions taken if the inoperable [PSW] subsystem results in an inoperable DG subsystem ~~or RHR shutdown cooling subsystem, respectively~~. This is in accordance with LCO 3.0.6 and ensures the proper actions are taken for these components.

Insert ACTION B

C.1 and C.2

If the [PSW] subsystem cannot be restored to OPERABLE status within the associated Completion Time, or both [PSW] subsystems are inoperable ~~for reasons other than Condition B and [Condition C]~~, [or the [UHS] is determined inoperable ~~for reasons other than Condition C~~] the ~~unit~~ plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the ~~unit~~ plant must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~unit~~ plant conditions from full

(continued)

Revision C

DB3 Insert ACTION B

B.1

With one division of deicing heaters inoperable, the deicing heaters must be restored to OPERABLE status within 7 days. With the plant in this condition, the remaining OPERABLE division of deicing heaters is adequate to perform the required function. However, the overall reliability of the deicing heaters is reduced.

The 7 day Completion Time is based on the redundant capabilities afforded by the OPERABLE division of deicing heaters, the low probability of an accident occurring during this time period, and is consistent with the allowed Completion Time for restoring an inoperable EDG subsystem.

BASES

DB3

SURVEILLANCE REQUIREMENTS

~~SR 3.7.2.4 (continued)~~

~~significant degradation of the cooling tower fans occurring between surveillances.~~

Insert SRs CLB3

SR 3.7.2.~~5~~ 4 DB3 CLB3

E

Verifying the correct alignment for each manual, power operated, and automatic valve in each [PSW] subsystem flow path provides assurance that the proper flow paths will exist for [PSW] operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be automatically realigned to its accident position within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. DB1 E PA1

This SR is modified by a Note indicating that isolation of the [PSW] System to components or systems may render those components or systems inoperable, but does not necessarily affect the OPERABILITY of the [PSW] System. As such, when all [PSW] pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the [PSW] System ~~is~~ may still be considered OPERABLE. E DB1 E PA1

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.2.~~6~~ 7 DB3 CLB3 DB5

Insert from next page DB5

~~This SR verifies that the automatic isolation valves of the [PSW] System will automatically switch to the safety or emergency position to provide cooling water exclusively to the safety related equipment during an accident event.~~ This is demonstrated by the use of an actual or simulated

(continued)

Revision J

CLB3 INSERT SRs

SR 3.7.2.3, SR 3.7.2.5, and SR 3.7.2.6

These SRs are modified by a NOTE indicating that these SRs are not required to be met if lake temperature is > 37°F. Industry experience has shown that frazil ice will not adhere to the bar racks that are above freezing temperatures. Therefore at these elevated temperatures, blockage of the intake tunnel is unlikely and the deicing heaters are not required to be OPERABLE.

Verification of the required deicing feeder current in SR 3.7.2.3 and the required deicing heater power in SR 3.7.2.5 will help ensure that adequate heat is being provided at the bar racks to help ensure that frazil ice does not adhere to them. Verification of the required deicing heater resistance to ground in SR 3.7.2.6 is performed to monitor long term degradation of the cable and heater insulations. SR 3.7.2.3 can be performed by measuring the current in all three phases of the feeder cables to each division and ensuring the total current is within limits to confirm that at least 18 deicing heaters are OPERABLE in each division. SR 3.7.2.5 is performed to verify that at least 18 deicing heaters in each division are each dissipating at least 1670 watts. The 7 day Frequency of SR 3.7.2.3 and the 6 month Frequency of SR 3.7.2.5 is based on operating experience that shows the heaters are reliable. The 12 month Frequency of SR 3.7.2.6 has shown that the components usually pass the SR when performed at the 12 month Frequency. Therefore, this Frequency is considered to be acceptable from a reliability standpoint.

associated with each EDG. In addition, the proper positioning of the ESW supply header isolation valves and the ESW minimum flow valves, upon actual or simulated ESW lockout matrix logic actuation must be demonstrated in this SR.

DB5

BASES

7 DB3

SURVEILLANCE REQUIREMENTS

SR 3.7.2.~~6~~ (continued)

initiation signal. This SR ~~also~~ verifies the automatic start capability ~~of one~~ of the ~~two~~ ~~PSW~~ pumps in each subsystem.

DB6 DB1 E CLB2 24

move to previous page DB5

Operating experience has shown that these components usually pass the SR when performed at the ~~18~~ month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

1. UFSAR, Section 9.7.1 DB9

REFERENCES

2 PA2 5 DB8
1. ~~U~~FSAR, Chapter ~~10~~.

PA2 14
3 ~~2~~. ~~U~~FSAR, Chapter ~~6~~.

DB9

4. 10 CFR 50.36 (c)(2)(ii). X1

The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.7.? overlaps this Surveillance to provide complete testing of the assumed safety function.

1 J

1 J

PA3

DB5

ESW will not be supplied to the Reactor Building Closed Loop Cooling Water System during the performance of this test to avoid contaminating this system with lake water

Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.2 - EMERGENCY SERVICE WATER (ESW) SYSTEM AND ULTIMATE HEAT SINK (UHS)

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The ITS 3.7.2 ACTIONS A Completion Time of 7 days is consistent with the current licensing basis (CTS 3.11.D.2) and with the Completion Time of an inoperable emergency diesel generator subsystem in ITS 3.8.1.

CLB2 The brackets have been removed and the proper value included. The ITS SR 3.7.2.4 Frequency of 24 months is consistent with the current licensing basis (CTS 4.11.D.1.a).

CLB3 ITS SRs 3.7.2.3, 3.7.2.5 and 3.7.2.6 have been added consistent with CTS 4.11.E. The Bases have been revised to revised to reflect the addition of these requirements. Subsequent SRs have also been renumbered, as applicable.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Changes have been made to be consistent with a change made to the Specification.

PA2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA3 Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific information/value has been provided.

DB2 ISTS 3.7.2 ACTIONS A and B have been deleted since each ESW subsystem at JAFNPP has only one pump. Subsequent ACTIONS have been renumbered and modified, as applicable.

DB3 ISTS 3.7.2 ACTION C and ISTS SRs 3.7.2.1 and 3.7.2.4 are being deleted because the design of JAFNPP Emergency Service Water System does not include cooling towers. However, ACTION B has been added to cover the condition where one division of required deicing heaters is inoperable. Subsequent ACTIONS and SRs have also been renumbered and modified, as applicable.

3.7 PLANT SYSTEMS

3.7.2 Emergency Service Water (ESW) System and Ultimate Heat Sink (UHS)

LCO 3.7.2 Two ESW subsystems and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ESW subsystem inoperable.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources – Operating," for emergency diesel generator subsystem made inoperable by ESW. -----------------------------	
	A.1 Restore the ESW subsystem to OPERABLE status.	7 days
B. One division of required deicing heaters inoperable. AND UHS temperature $\leq 37°F$.	B.1 Restore the division of required deicing heaters to OPERABLE status.	7 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify the water level in the ESW pump screenwell is ≥ 236.5 ft mean sea level.	24 hours
SR 3.7.2.2	Verify the average water temperature of UHS is ≤ 85°F.	24 hours
SR 3.7.2.3	------------------NOTE------------------ Not required to be met if UHS temperature is > 37°F. --- Verify the required deicing heater feeder current is within limits for each division of deicing heaters.	7 days
SR 3.7.2.4	------------------NOTE------------------ Isolation of flow to individual components does not necessarily render ESW System inoperable. --- Verify each ESW subsystem manual, power operated, and automatic valve in the flow paths servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.2 Emergency Service Water (ESW) System and Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The ESW System is designed to provide cooling water for the removal of heat from equipment, such as the emergency diesel generators (EDGs), electric bay coolers, crescent area coolers, cable tunnel/switchgear room coolers and control room and relay room air handling units, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. Upon receipt of a loss of offsite power or loss of coolant accident (LOCA) signal, the EDGs will start which in turn starts the associated ESW pump. Each ESW pump will automatically pump to the associated EDG cooler. The remaining ESW loads will be automatically cooled when the associated ESW supply header isolation valve opens and the associated ESW minimum flow valve closes. This occurs when the ESW lockout matrix logic actuates upon low reactor building closed loop cooling water pump discharge pressure. This logic is discussed in LCO 3.3.7.3, "Emergency Service Water (ESW) System Instrumentation". In addition, the ESW pumps will automatically start in response to the ESW lockout matrix logic. However, this function is not required for safe reactor shutdown since the ESW pumps will start when any associated EDG starts.

The ESW System consists of the UHS and two independent and redundant subsystems. Each of the two ESW subsystems is made up of a header, one 3700 gpm pump, a suction source, valves, piping and associated instrumentation. The two subsystems are separated from each other so failure of one subsystem will not affect the OPERABILITY of the other system. The ESW System is described in UFSAR, Section 9.7.1 (Ref. 1).

Cooling water flows from Lake Ontario (UHS) through the intake tunnel to the screenwell where the water is pumped by the ESW pumps to components through the two main headers. After removing heat from the components, the water is discharged to the discharge tunnel where it returns to Lake Ontario.

The lake intake structure is a reinforced concrete structure sitting on the lake bottom at a distance of approximately 900 ft from the shoreline in approximately 25 ft of water. 1

(continued)

BASES

BACKGROUND (continued)

The top surface of the intake structure is at the 233 ft elevation (above sea level), which is approximately 10 ft below the historically lowest monthly mean lake level. The intake is a roofed structure which draws water in through side openings that are protected with bar racks spaced at 1 ft centers to block the entrance of large debris. This results in water being taken in at lower levels and prevents the formation of vortices at the surface, thus minimizing the possibility of floating ice being drawn down from the surface. The side intake area of approximately 8 ft by 70 ft, less bar rack area, provides a net clear area of 552 ft^2. During normal operation, with a maximum nominal operating flow of 388,600 gpm from three circulating water pumps and two normal service water pumps, the average intake velocity is approximately 1.6 ft per second. However, during safe shutdown conditions with only two Residual Heat Removal Service Water (RHRSW) pumps and one ESW pump in operation, the maximum nominal flow is reduced to 10,000 gpm, corresponding to an average intake velocity of 0.04 ft per second.

The formation of frazil ice on the steel bar racks at the intake structure openings is common in northern climates. This kind of ice is formed when meteorological conditions are such that the water is subcooled below its freezing point due to radiational cooling. Under these conditions, frazil ice can form on intake bar racks or spongy masses of this ice, formed in other parts of the lake and carried past an intake by wind-driven currents, can adhere to the bar racks. Sufficient transport velocity exists to move buoyant frazil ice from the lake surface to the intake structure during normal operation, but not under safe shutdown conditions. If ice formation does occur on the bar racks during normal operation, sufficient local erosion velocities will develop to limit total ice accumulation such that the remaining net clear intake area would be sufficient to meet required safe shutdown flows. In an effort to suppress the formation of frazil ice on the bar racks, each of the 88 rack bars is heated by a deicing heater. Each deicing heater is rated at 1670 watts and is normally energized. Forty four heaters are powered by one division while the remaining 44 heaters are powered by the other division.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Since Lake Ontario is the UHS, sufficient water inventory is available for all ESW System post LOCA cooling requirements for a 30 day period. The OPERABILITY of the ESW System is assumed in evaluations of the equipment required for safe reactor shutdown presented in the UFSAR, Chapters 5 and 14 (Refs. 2 and 3, respectively). These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The ability of the ESW System to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analyses evaluated in References 2 and 3. The ability to provide onsite emergency AC power is dependent on the ability of the ESW System to cool the EDGs. The long term cooling capability of RHR and core spray pumps is dependent on the capability of the ESW System to provide cooling to the EDGs as well as the crescent area coolers.

The ESW System, together with the UHS, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The ESW subsystems are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one subsystem of ESW is required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, two subsystems of ESW must be OPERABLE. At least one subsystem will operate, if the worst single active failure occurs coincident with the loss of offsite power.

A subsystem is considered OPERABLE when it has an OPERABLE UHS, one OPERABLE pump, and an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the appropriate equipment. OPERABILITY of equipment cooled by the ESW System is based on heat transfer, not flow rates; OPERABILITY of the ESW pumps is based on measured performance remaining within allowable IST Program acceptance criteria. J

The OPERABILITY of the UHS is based on having a minimum water level in the screenwell of 236.5 ft mean sea level and a maximum water temperature of 85°F. With UHS temperature (J

(continued)

BASES

LCO (continued)

$\leq$ 37°F, conditions become increasingly favorable for the formation of frazil ice on the intake structure bar racks during normal operation. Therefore, in an effort to suppress the formation of frazil ice on the intake structure bar racks, at least 18 out of the 44 deicing heaters (each heater producing 1670 watts) in each electrical division are maintained OPERABLE whenever UHS temperature is $\leq$ 37°F.

J

The isolation of the ESW System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the ESW System.

APPLICABILITY

In MODES 1, 2, and 3, the ESW System and UHS are required to be OPERABLE to support OPERABILITY of the equipment serviced by the ESW System. Therefore, the ESW System and UHS are required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the ESW System and UHS are determined by the systems they support and therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. Thus, LCO 3.7.4, "Control Room AC System," and LCO 3.8.2, "AC Sources - Shutdown," which require the ESW System to be OPERABLE, will govern ESW System operation in MODES 4 and 5.

J

ACTIONS

A.1

With one ESW subsystem inoperable, the ESW subsystem must be restored to OPERABLE status within 7 days. With the plant in this condition, the remaining OPERABLE ESW subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single active component failure in the OPERABLE ESW subsystem could result in loss of ESW function.

The 7 day Completion Time is based on the redundant ESW System capabilities afforded by the OPERABLE subsystem, the low probability of an accident occurring during this time period, and is consistent with the allowed Completion Time for restoring an inoperable EDG subsystem.

(continued)

BASES

ACTIONS

A.1 (continued)

Required Action A.1 is modified by a Note indicating that the applicable Conditions of LCO 3.8.1, "AC Sources - Operating," be entered and Required Actions taken if the inoperable ESW subsystem results in an inoperable EDG subsystem. This is in accordance with LCO 3.0.6 and ensures the proper actions are taken for this component.

B.1

With one division of deicing heaters inoperable, the deicing heaters must be restored to OPERABLE status within 7 days. With the plant in this condition, the remaining OPERABLE division of deicing heaters is adequate to perform the required function. However, the overall reliability of the deicing heaters is reduced.

The 7 day Completion Time is based on the redundant capabilities afforded by the OPERABLE division of deicing heaters, the low probability of an accident occurring during this time period, and is consistent with the allowed Completion Time for restoring an inoperable EDG subsystem.

C.1 and C.2

If the ESW subsystem cannot be restored to OPERABLE status within the associated Completion Time, or both ESW subsystems are inoperable, or the UHS is determined inoperable the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies the water level in the screenwell to be sufficient for the proper operation of the ESW and RHRSW pumps (net positive suction head and pump vortexing are considered in determining this limit). The 24 hour

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.2.2

Verification of the UHS temperature ensures that the heat removal capability of the ESW System is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.2.3, SR 3.7.2.5, and SR 3.7.2.6

These SRs are modified by a NOTE indicating that these SRs are not required to be met if UHS temperature is > 37°F. Industry experience has shown that frazil ice will not adhere to the bar racks that are above freezing temperatures. Therefore at these elevated temperatures, blockage of the intake is unlikely and the deicing heaters are not required to be OPERABLE.

Verification of the required deicing feeder current in SR 3.7.2.3 and the required deicing heater power in SR 3.7.2.5 will help ensure that adequate heat is being provided at the bar racks to help ensure that frazil ice does not adhere to them. Verification of the required deicing heater resistance to ground in SR 3.7.2.6 is performed to monitor long term degradation of the cable and heater insulations. SR 3.7.2.3 can be performed by measuring the current in all three phases of the feeder cables to each division and ensuring the total current is within limits to confirm that at least 18 deicing heaters are OPERABLE in each division. SR 3.7.2.5 is performed to verify that at least 18 deicing heaters in each division are each dissipating at least 1670 watts. The 7 day Frequency of SR 3.7.2.3 and the 6 month Frequency of SR 3.7.2.5 is based on operating experience that shows the heaters are reliable. The 12 month Frequency of SR 3.7.2.6 has shown that the components usually pass the SR when performed at the 12 month Frequency. Therefore, this Frequency is considered to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.4

Verifying the correct alignment for each manual, power operated, and automatic valve in each ESW subsystem flow path provides assurance that the proper flow paths will exist for ESW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be automatically realigned to its accident position within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This SR is modified by a Note indicating that isolation of the ESW System to components or systems may render those components or systems inoperable, but does not necessarily affect the OPERABILITY of the ESW System. As such, when all ESW pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the ESW System may still be considered OPERABLE.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.2.7

This SR verifies the automatic start capability of the ESW pump in each subsystem. This is demonstrated by the use of an actual or simulated initiation signal associated with each EDG. In addition, the proper positioning of the ESW supply header isolation valves and the ESW minimum flow valves, upon actual or simulated ESW lockout matrix logic actuation, must be demonstrated in this SR. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.7.3 overlaps this Surveillance to provide complete testing of the assumed safety function. ESW will not be supplied to the Reactor Building Closed Loop Cooling System during the performance of this test to avoid contaminating this system with lake water.

(continued)

Specification 3.7.3

A1

3.11 (cont'd)

3. The control room emergency ventilation system shall not be out of service for a period exceeding 3 days during normal reactor operation or refueling operations. In the event that the system is not returned to service within 3 days, ~~the reactor shall be in cold shutdown within 24 hours and any handling of irradiated fuel, core alterations, and operations~~ with a potential for draining the reactor vessel shall be suspended as soon as practicable

[ACTION E]

Enter LCO 3.0.3

M4

[ACTION F]

add ACTION F Note

M6

4. Not Used

add STAGGERED TEST BASIS

L1

M5

Verify each CREVAS subsystem can maintain a positive pressure of ≥ 0.125 in. water gauge relative to the atmosphere and turbine building during the isolation mode of operation at a flow rate ≥ 900 scfm and ≤ 1100 scfm.

[SR 3.7.3.2]

B. DELETED

C. Battery Room Ventilation

Battery room ventilation shall be operable on a continuous basis whenever specification 3.9.E is required to be satisfied.

1. From and after the date that one of the battery room ventilation systems is made or found to be inoperable, its associated battery shall be considered to be inoperable for purposes of specification 3.9.E.

4.11 (cont'd)

See ITS 3.3.7.1

3. Operability of the main control room air intake radiation monitor shall be tested once/3 months.

LB1

4. Temperature transmitters and differential pressure switches shall be calibrated once per 24 months.

5. Main control room emergency ventilation air supply system capacity shall be tested once every 18 months to assure that it is ± 10% of the design value of 1000 cfm.

B. DELETED

C. Battery Room Ventilation

Battery room ventilation equipment shall be demonstrated operable once/week.

1. When it is determined that one battery room ventilation system is inoperable, the remaining ventilation system shall be verified operable and daily thereafter.

2. Temperature transmitters and differential pressure switches shall be calibrated once per 24 months.

See CTS 3/4.11.C in this Section

A

Amendment No. ~~48, 82, 126, 134, 148, 156, 231, 233~~, 269

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