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NL-11-087

September 16, 2011

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

SUBJECT: Proposed License Amendment Regarding Travelers – TSTF-479-A and TSTF-497-A
Indian Point Nuclear Generating Unit Nos. 2 and 3
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

REFERENCES
1. TSTF-479-A, Revision 0, "Changes to Reflect Revision to 10 CFR 50.55a," dated December 19, 2005
2. TSTF-497-A, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less," dated August 28, 2008.

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Northeast, (Entergy) hereby requests a License Amendment to operating license DPR-26, Docket No. 50-247 Indian Point Nuclear Generating Unit 2 (IP-2) and to operating license DPR-64, Docket No. 50-286 Indian Point Nuclear Generating Unit 3 (IP-3). Currently, Technical Specifications (TS) for the Inservice Testing Program contains references to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI as the source of requirements for the inservice testing (IST) of ASME Code Class 1, 2, and 3 pumps and valves. The proposed change deletes the references to Section XI of the Code and incorporates references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code). The proposed change also indicates that the allowance for a 25% extension of surveillance intervals may be applied to accelerated frequencies utilized in the IST Program. The proposed changes are consistent with Technical Specification Task Force (TSTF) Technical Change Travelers 479-A (Reference 1) and 497-A (Reference 2). These travelers revised the Improved Standard Technical Specifications. The changes are also consistent with the implementation of the IP-2 and IP-3 fourth (4th) 10-year interval IST program in accordance with the requirements of 10 CFR 50.55a(f). The 4th 10-year interval began March of 2007 for IP-2 and July of 2009 for IP-3.

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Entergy has evaluated the proposed changes in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and has determined that the changes involve no significant hazards considerations, as described in Attachment 1. The marked up pages showing the proposed Technical Specification changes are provided in Attachment 2. The proposed changes for the associated Bases changes are provided in Attachment 3 for information. A copy of this application and the associated attachments are being submitted to the designated New York State official in accordance with 10 CFR 50.91.

There are no new commitments identified in this submittal. If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 9/16/2011.

Sincerely,

J. Pollock

JP/mb

Attachments:
1. Analysis of Proposed Technical Specifications Changes Regarding Travelers – TSTF-479-A and TSTF-497-A
2. Markup of Technical Specifications Pages for Proposed Changes Regarding Travelers – TSTF-479-A and TSTF-497-A
3. Markup of Technical Specifications Bases Associated with the Proposed Changes Regarding Travelers – TSTF-479-A and TSTF-497-A

cc: Mr. John P. Boska, Senior Project Manager, NRC NRR DORL
Mr. William Dean, Regional Administrator, NRC Region 1
NRC Resident Inspectors
Mr. Francis J. Murray, Jr., President and CEO, NYSERDA
Mr. Paul Eddy, New York State Dept. of Public Service

ATTACHMENT 1 TO NL-11-087

Analysis of Proposed Technical Specifications Changes Regarding Travelers – TSTF-479-A and TSTF-497-A

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNITS 2 AND 3
DOCKET NOS. 50-247 & 50-286

1.0 DESCRIPTION

Entergy Nuclear Northeast, (Entergy) requests an amendment to Operating Licenses DPR-26 for the Indian Point Nuclear Generating Unit 2 (IP-2) and DPR-64 for the Indian Point Nuclear Generating Unit 3 (IP-3). The proposed amendment revises the Technical Specification (TS), Administrative Controls, "Inservice Testing Program" for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The amendment also establishes the 25 percent extension for accelerated frequencies used in the Inservice Test (IST) Program.

Entergy proposes these changes based on NRC-approved Technical Specification Task Force (TSTF) TSTF-479-A, "Changes to Reflect Revision of 10 CFR 50.55a," Revision 0, and TSTF-497-A, "Limit Inservice Testing Program [Surveillance Requirement] SR 3.0.2 Application to Frequencies of Two Years or Less," Revision 0.

2.0 PROPOSED CHANGE

The proposed changes will revise IP-2, TS 5.5.6, Inservice Testing (IST) Program and IP-3, TS 5.5.7, Inservice Testing (IST) Program as follows:

Change:

> Section XI of the ASME Boiler and Pressure Vessel Code

To

> ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code)

Change:

> The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities.

To

> The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities.

Various sections of TS Bases will be revised in accordance with IP-2, TS 5.5.12 and IP-3, TS 5.5.13, Technical Specification (TS) Bases Control Program, after the incorporation of this request into the IP-2 and IP-3 TS.

3.0 BACKGROUND

In 1990, the ASME published the initial edition of the ASME OM Code which establishes rules for IST of pumps and valves. The ASME intended that the ASME OM Code replace Section XI of the Boiler and Pressure Vessel Code for IST of pumps and valves.

On December 2, 2004, the TSTF submitted to the NRC TSTF-479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a." The proposed change revised the IST Program TS located in Chapter 5 of the Improved Standardized Technical Specifications (ISTS) to reflect the latest NRC-approved version of the ASME Code. TSTF-479 also revised paragraph (b) of the IST Program TS to state, "The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified in the IST Program for performing inservice testing activities."

In letter dated December 6, 2005, the NRC approved TSTF-479 as an administrative change to the ISTS NUREGs. TSTF-479-A was incorporated into Revision 3.1 of the ISTS NUREGs.

At the February 23, 2006, meeting between the NRC and the TSTF, members of the Component Branch of the NRC stated that TSTF-479 did not provide an adequate justification for applying SR 3.0.2 to frequencies specified in the IST Program TS as greater than 2 years and the NRC would not approve plant-specific amendments based on TSTF-479-A incorporating this change without further justification. The NRC stated that they would accept applying SR 3.0.2 to IST frequencies not listed in the IST Program TS table provided that those frequencies are specified in the IST Program TS as 2 years or less.

In response, TSTF-497-A was developed as an administrative change to the ISTS NUREGs to reflect the NRC position. TSTF-497-A revises paragraph (b) of the IST program in the ISTS to state, "The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities."

4.0 TECHNICAL ANALYSIS

The purpose of the IST program is to assess the operational readiness of pumps and valves, to detect degradation that might affect component operability, and to maintain safety margins with provisions for increased surveillance and corrective action. NRC regulation, 10 CFR 50.55a, defines the requirements for applying industry codes to each licensed nuclear powered facility. Licensees are required by 10 CFR 50.55a(f)(4)(i) to initially prepare programs to perform IST of certain ASME Section III, Code Class 1, 2, and 3 pumps and valves during the initial 120-month interval. The regulations require that programs be developed utilizing the latest edition and addenda incorporated into paragraph (b) of 10 CFR 50.55a on the date 12 months prior to the date of issuance of the operating license subject to the limitations and modification identified in paragraph (b).

NRC regulations also require that the IST programs be revised during successive 120-month intervals to comply with the latest edition and addenda of the Code incorporated by reference in paragraph (b) 12 months prior to the start of the interval.

Section XI of the ASME Code has been revised on a continuing basis over the years to provide updated requirements for the inservice inspection and IST of components. Until 1990, the ASME Code requirements addressing the IST of pumps and valves were contained in Section XI, Subsections IWP (pumps) and IWV (valves). In 1990, the ASME published the initial edition of the OM Code that provides the rules for the IST of pumps and valves. Since the establishment of the 1990 Edition of the OM Code, the rules for the IST of pumps are no longer being updated in Section XI. Therefore, the IP-2 and IP-3 TS are revised to appropriately refer to the ASME OM Code, consistent with ISTS NUREGs. As identified in NRC SECY-99-017 dated January 13, 1999, the NRC has generally considered the evolution of the ASME Code to result in a net improvement in the measures for inspecting piping and components and testing pumps and valves.

In addition to changes related to application of the ASME OM Code above, the TS IST Program is revised to indicate that the provisions of SR 3.0.2 are applicable to other IST frequencies that are not specified in the Program. The IST Program TS may have frequencies for testing that are based on risk and do not conform to the standard testing frequencies specified in the TS. For example, an IST Program may use ASME Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light-Water Reactor Plants," in lieu of stroke time testing. The Frequency of the Surveillance may be determined through a mix of risk informed and performance based means in accordance with the IST program. This is consistent with the guidance in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," which indicates that the 25% extension of the interval specified in the Frequency would apply to increased frequencies the same as it applies to regular frequencies. If a test interval is specified in 10 CFR 50.55a, the TS SR 3.0.2 Bases indicates that the requirement of the regulation take precedence over the TS's.

At the February 23, 2006, meeting between the NRC and the TSTF, members of the Component Branch of the NRC stated that TSTF-479 did not provide an adequate justification for applying SR 3.0.2 to frequencies specified in the IST program as greater than 2 years and the NRC would not approve plant-specific amendments based on TSTF-479 incorporating this change without further justification. After consideration, the TSTF declined to develop a technical justification for applying SR 3.0.2 to IST frequencies specified as greater than 2 years at the time due to inadequate cost benefit. As a result, TSTF-497-A was developed as an administrative change to the ISTS NUREGs which modifies the IST program, paragraph (b), to remove the provisions that were not deemed by the NRC to be adequately justified in TSTF-479.

Based on the above, the proposed change is consistent with the ISTS NUREGs, TSTF-479-A, TSTF-497-A , and the previous NRC request regarding application of SR 3.0.2. Therefore, Entergy considers the proposed change to have no significant increase in the probability or consequences of an accident because the changes correct the reference in the TS to the proper ASME code and extend the 25% extension of the specified test intervals to accelerated intervals not exceeding two years in the same manner as the extension is applied to standard intervals. The proposed changes do not create the possibility of a new or different type of accident than previously evaluated since the changes do not involve addition or removal of any equipment to the facility or change plant operation. The proposed changes do not involve a significant reduction in margin of safety because the safety function of the affected components are maintained in the same manner and the 25% extension to accelerated intervals is not providing any reduction greater than the current provisions do for standard intervals.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

A change is proposed to the IP-2, TS 5.5.6 and IP-3, TS 5.5.7 Inservice Testing (IST) Program to adopt the American Society of Mechanical Engineers (ASME) Operations and Maintenance (OM) Code for IST of valves and pumps. The proposed change includes application of the allowances provided by TS Surveillance Requirement (SR) 3.0.2 for IST SR frequencies of 2 years or less. Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   The proposed change revises IP-2, TS 5.5.6 and IP-3, TS 5.5.7, Inservice Testing Program, for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

   The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. The proposed change does not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, this proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed change revises IP-2, TS 5.5.6 and IP-3, TS 5.5.7, Inservice Testing Program, for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure. Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

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

   Response: No

   The proposed change revises IP-2, TS 5.5.6 and IP-3, TS 5.5.7, Inservice Testing Program, for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. The safety function of the affected pumps and valves are maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

5.2 Applicable Regulatory Requirements / Criteria

NRC regulation, 10 CFR 50.55a, defines the requirements for applying industry codes to each licensed nuclear powered facility. Licensees are required by 10 CFR 50.55a(f)(4)(i) to initially prepare programs to perform inservice testing of certain ASME Section III, Code Class 1, 2, and 3 pumps and valves during the initial 120-month interval. The regulations require that programs be developed utilizing the latest edition and addenda incorporated into paragraph (b) of 10 CFR 50.55a on the date 12 months prior to the date of issuance of the operating license subject to the limitations and modification identified in paragraph (b).

This TS change will ensure the above regulation continues to be met by application of the ASME OM Code, which replaced Section XI of the Boiler and Pressure Vessel Code for Inservice Testing (IST) of pumps and valves in 1990. Therefore, based on the considerations discussed above:

1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;

2) Such activities will be conducted in compliance with the Commission's regulations; and

3) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. 10 CFR 50.55a
2. SECY-99-017, "Proposed Amendment to 10 CFR 50.55a"
3. NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants"

ATTACHMENT 2 TO NL-11-087

Markup of Technical Specifications Pages for Proposed Changes Regarding Travelers – TSTF-479-A and TSTF-497-A

Bold, italics for added text ~~Strikeout~~ for deleted text

AFFECTED PAGES (IP2)

5.5 - 5
5.5 - 6

AFFECTED PAGES (IP3)

5.0 - 12

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5.5 Programs and Manuals

5.5.3 Radioactive Effluent Controls Program (continued)

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.4 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR, Section 4.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.5 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel using ultrasonic methods. The program shall include inspection frequencies and acceptance criteria. The inspection frequency will ensure that each reactor coolant pump flywheel is inspected at 20-year intervals.

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies ~~specified in Section XI of the ASME Boiler and Pressure Vessel Code~~ ***applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code)*** and applicable Addenda as follows:

ASME ***OM*** ~~Boiler and Pressure Vessel~~ Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies ***and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program*** for performing inservice testing activities,

c. The provisions of SR 3.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME ***OM*** ~~Boiler and Pressure Vessel~~ Code shall be construed to supersede the requirements of any TS.

5.5.7 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads

5.5 Programs and Manuals

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies ***applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code)*** ~~specified in Section XI of the ASME Boiler and Pressure Vessel Code~~ and applicable Addenda as follows:

ASME *OM* ~~Boiler and Pressure Vessel~~ Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies ***and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program*** for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME *OM* ~~Boiler and Pressure Vessel~~ Code shall be construed to supersede the requirements of any TS.

(continued)

ATTACHMENT 3 TO NL-11-087

Markup of Technical Specifications Bases Associated with the Proposed Changes Regarding Travelers – TSTF-479-A and TSTF-497-A

Bold, italics for added text ~~Strikeout~~ for deleted text

AFFECTED PAGES (IP2)	AFFECTED PAGES (IP3)
3.4.10 - 4	3.4.10 - 5
3.4.11 - 7	3.4.11 - 7
3.4.14 - 6	3.4.14 - 8
3.4.14 - 8	3.4.14 - 10
3.5.2 - 11	3.5.2 - 10
3.6.6 - 9	3.5.2 - 11
3.6.6 - 11	3.6.6 - 11
3.7.1 – 5	3.6.6 - 13
3.7.1 – 6	3.7.1 – 5
3.7.2 – 7	3.7.1 – 6
3.7.2 – 8	3.7.2 – 9
3.7.3 – 7	3.7.2 – 10
3.7.5 – 8	3.7.3 – 7
3.7.5 – 9	3.7.5 – 8
3.8.1 – 22	3.7.5 – 10

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BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ Low Temperature Overpressure Protection (LTOP) Applicability temperature specified in LCO 3.4.12 within 24 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. With any RCS cold leg temperatures at or below the Low Temperature Overpressure (LTOP) Applicability temperature specified in LCO 3.4.12, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements ~~of Section XI~~ of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

2. UFSAR, Chapter 14.1.

3. WCAP-7769, Rev. 1, June 1972.

~~4. ASME, Boiler and Pressure Vessel Code, Section XI.~~

4. ***ASME code for Operation and Maintenance of Nuclear Power Plants.***

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

This SR requires block valve cycling to verify that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code~~, Section XI~~ (Ref. 3).

This SR is modified by two Notes. Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable. Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.

The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2.

REFERENCES

1. UFSAR, Sections 4.2 and 4.3.

2. UFSAR, Section 14.

~~3. ASME, Boiler and Pressure Vessel Code, Section XI.~~

3. ASME code for Operation and Maintenance of Nuclear Power Plants.

BASES

ACTIONS (continued)

C.1

If one or both RCS boundary valves 730 and 731 are not closed and de-energized, a single failure of the remaining barrier has the potential to cause a LOCA in which the containment and containment safeguards radionuclide protective barriers are bypassed (Ref. 10). Therefore, action must be taken to ensure that RCS boundary valves 730 and 731 are closed and de-energized. The Completion Time of 24 hours is acceptable because one or both valves are closed and provide an adequate RCS pressure boundary and the low probability of an event that could cause the failure of both RCS pressure boundary valves during this period.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 5 gpm maximum applies to each valve. However, if the leakage is greater than 1.0 gpm and the leak test indicates that there is significant deterioration from the previous leak test, then the results are unacceptable because of the adverse trend. Significant deterioration is indicated when the leakage is greater than the results of the previous test plus one-half of the margin following the previous test (i.e., margin following previous test is the 5.0 gpm limit minus the results of the previous test). Leakage limit acceptance criteria is based on the leakage rate that would exist when the RCS is at normal operating pressure. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 24 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code~~, Section XI~~ (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply

BASES

REFERENCES (continued)

4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.

5. NUREG-0677, May 1980.

6. UFSAR, Section 5.2.

~~7. ASME, Boiler and Pressure Vessel Code, Section XI.~~

7. ASME code for Operation and Maintenance of Nuclear Power Plants.

8. 10 CFR 50.55a(g).

9. Generic Letter 87-006, Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves.

10. WCAP-11736-A, Residual Heat Removal System Autoclosure Interlock (ACI) Removal Report.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2 (continued)

This Surveillance 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 administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by ~~Section XI of~~ the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program~~, which encompasses Section XI~~ of the ASME Code. ~~Section XI of t~~The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.4 and SR 3.5.2.5

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. Note that the Containment Recirculation system is a manually initiated system and is not included as part of this SR. Additionally, this Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray 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. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

Starting each FCU from the control room and operating each containment FCU fan for ≥ 15 minutes ensures that all FCUs are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering FCUs are operated during normal plant operation, the known reliability of the fan units and controls, the redundancy available, and the low probability of significant degradation of the FCUs occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying flow rate to each cooling unit is ≥ 1600 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The 92 day Frequency was developed considering quarterly swapping of the essential and non-essential SW headers, the known reliability of the Service Water System, the redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref. 6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.9

This SR verifies that minimum air flow through each FCU equals the air flow assumed in the accident analysis for heat removal from the containment. The 24 month Frequency is based on engineering judgment.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. 10 CFR 50, Appendix K.
3. UFSAR, Section 6.3.
4. UFSAR, Section 6.4.
5. UFSAR, Section 14.3.
6. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~
6. ***ASME code for Operation and Maintenance of Nuclear Power Plants.***
7. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 211 to Facility Operating License No. DPR-26, July 27, 2000.

BASES

ACTIONS (continued)

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Protection System Instrumentation," already establish a trip setpoint lower than that required by this LCO.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

C.1 and C.2

If the Required Actions are not completed within the associated Completion Time, or if one or more steam generators have ≥ 4 inoperable MSSVs, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code~~, Section XI~~ (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination,

b. Seat tightness determination,

c. Setpoint pressure determination (lift setting), and

d. Compliance with owner's seat tightness criteria.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested *in situ* at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. UFSAR, Section 10.2.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. UFSAR, Section 14.
4. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~
4. ***ASME code for Operation and Maintenance of Nuclear Power Plants.***
5. ANSI/ASME OM-1-1987.
6. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5.0 seconds. The MSIV isolation time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power because even a part stroke causes a turbine trip. As the MSIVs are not tested at power, they are exempt from the ASME Code~~, Section XI~~ (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MSIV testing is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at this Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

SR 3.7.2.3

Each MSCV must be inspected to ensure that it closes properly. This ensures that the safety analysis assumptions are met. The Frequency of this SR is based on Inservice Testing Program requirements and corresponds to the expected refueling cycle.

BASES

REFERENCES

1. UFSAR, Section 10.2.

2. UFSAR, Section 14.2.

3. 10 CFR 50.67.

4. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 211 to Facility Operating License No. DPR-26, July 27, 2000.

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

5. ASME code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.2 requires verification that MBFP discharge valves and MBFP trips within the following limits:

MBFP discharge valves (BFD-2-21 and BFD-2-22) close in ≤ 60 seconds; and

MBFPs (BFP 21 and BFP 22) trip in ≤ 5 seconds.

These Surveillances are normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code~~, Section XI~~ (Ref. 3), quarterly stroke requirements during operation in MODES 1 and 2. Surveillance testing of the MFIVs (BFD-5, BFD-5-1, BFD-5-2, and BFD-5-3) is administratively controlled.

The Frequency for these SRs is in accordance with the Inservice Testing Program.

SR 3.7.3.3

This SR verifies that each MFRV, Low Flow FBV, and MBFP discharge valve will close and that each MBFP will trip on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Frequency for this SR is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.2.

2. UFSAR, Section 14.2.

~~3. ASME, Boiler and Pressure Vessel Code, Section XI.~~

3. ASME code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code~~, Section XI~~ (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test when SG pressure is < 600 psig.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage (i.e., unit at less than or equal to 97% power and in preparation for main generator breaker opening with no plans to raise power between the time of the surveillance and breaker open) and the potential for an unplanned transient if the Surveillance were performed with the reactor at full power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during

BASES

SURVEILLANCE REQUIREMENTS (continued)

startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained.

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is aligned and operated as necessary to maintain SG water level.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an actuation signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is operated as necessary to maintain SG water level and the autostart function is not required. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allowing the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System. OPERABILITY (i.e., the intended safety function) continues to be maintained.

REFERENCES

1. UFSAR, Section 10.2.

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

2. ASME code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency for this SR is 92 days. The 92 day Frequency corresponds to the testing requirements for pumps as contained in the ASME Code~~, Section XI~~.

SR 3.8.1.7

Transfer of each offsite power supply from the 138 kV offsite circuit to the 13.8 kV offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The 24 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced.

SR 3.8.1.8

Verification that 6.9 kV buses 2 and 3 will auto transfer (dead fast transfer) from the Unit Auxiliary Transformer (the main generator) to 6.9 kV buses 5 and 6 (the offsite circuit) following a loss of voltage on 6.9 kV buses 2 and 3 is needed to confirm the OPERABILITY of a function assumed to operate to provide offsite power to safeguards power train 2A/3A following a trip of the main generator. (Note that when the main generator trips on over-frequency, the transfer is blocked by an over-frequency transfer interrupt circuit provided for bus protection of out of phase transfer.)

An actual demonstration of this feature requires the tripping the main generator while the reactor is at power with the main generator supplying 6.9 kV buses 2 and 3. Credit may be taken for planned plant trips or for unplanned events that satisfy this SR. Other than planned plant trips or unplanned events, Note 1 specifies that this SR is not normally performed in MODE 1 or 2 because performance of this SR could cause perturbations to

BASES

ACTIONS

B.1 and B.2 (continued)

and without challenging plant systems. With any of the RCS cold leg temperatures ≤ 330°F (i.e., when LCO 3.4.12 is applicable) overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements ~~of Section XI~~ of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

2. FSAR, Chapter 14.

3. WCAP-7769, Rev. 1, June 1972.

~~4. ASME, Boiler and Pressure Vessel Code, Section XI.~~

4. ASME code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code~~, Section XI~~ (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is important because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an inoperable PORV that is not capable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 7 days, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status.

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. Regulatory Guide 1.32, February 1977.

2. FSAR, Section 14.

~~3. ASME, Boiler and Pressure Vessel Code, Section XI.~~

3. ASME code for Operation and Maintenance of Nuclear Power Plants.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 24 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code~~, Section XI~~ (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within
24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

(continued)

BASES

REFERENCES (continued)

4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. FSAR Section 6.2.
7. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~
7. *ASME code for Operation and Maintenance of Nuclear Power Plants.*
8. 10 CFR 50.55a(g).
9. Generic Letter 87-006, Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves.
10. WCAP-11736-A, Residual Heat Removal System Autoclosure Interlock (ACI) Removal Report.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render more than one ECCS train inoperable. Securing these valves in position by removal of power or by key locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 5, that can disable the function of more than one ECCS train and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance 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 administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by ~~Section XI of~~ the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3

flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program~~, which encompasses Section XI~~ of the ASME Code. ~~Section XI of t~~The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.4 and SR 3.5.2.5

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. Note that the Containment Recirculation system is a manually initiated system and is not included as part of this SR. Additionally, this Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.6

Alignment of valves in the HHSI flow path is necessary for proper ECCS performance. These valves have stops to allow proper positioning and/or locking manual valve in the flow path for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow, and to allow proper positioning for restricting hot leg flow. Therefore, an improperly positioned valve could result in the inoperability of more than one injection flow path. The stops and/or the locked manual valves are set based on the results of the most recent ECCS operational flow test. Valves SI-856B, 856C, 856D, 856E, 856H, 856J, and 856K are not necessarily *used for flow balancing but can be used to trim system* resistance during flow balance testing. The stop positions are set to reflect their usage. The 24 month Frequency is based on the reasons stated in SR 3.5.2.4 and SR 3.5.2.5.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Operating each containment fan cooler unit for ≥ 15 minutes ensures that all fan cooler units are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 92 day Frequency was developed considering fan coolers are operated during normal plant operation, the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the containment fan cooler units occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying that the service water flow rate to each fan cooler unit is ≥ 1400 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The 92 day Frequency was developed considering the known reliability of the Cooling Water System, the redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref. 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.8 (continued)

the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.6.9

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. 10 CFR 50, Appendix A.
2. 10 CFR 50, Appendix K.
3. FSAR, Sections 6.3 and 6.4.
4. FSAR, Section 14.3 Table 14.3-56.
~~5. ASME, Boiler and Pressure Vessel Code, Section XI.~~
5. ASME code for Operation and Maintenance of Nuclear Power Plants.
6. WCAP - 16212P, Indian Point Nuclear Power Generating Unit No. 3 Stretch Power Uprate NSSS and BOP Licensing Report, June 2004.

BASES

ACTIONS (continued)

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code~~, Section XI~~ (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section 10.2.
2. ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition.
3. FSAR, Section 14.
4. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~
4. ***ASME code for Operation and Maintenance of Nuclear Power Plants.***
5. ANSI/ASME OM-1-1987.
6. Nuclear Safety Advisory Letter (NSAL) 94-001, Operation at Reduced Power Levels with Inoperable Main Steam Safety Valves.
7. Information Notice 94-60, Potential Overpressurization of Main Steam System.

BASES

ACTIONS

G.1 and G.2

If the MSIVs or MSCVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5.0 seconds on an actual or simulated actuation signal. The MSIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs are not tested at power because even a part stroke causes a turbine trip and valve closure. As the MSIVs are not tested at power, they are exempt from the ASME Code~~, Section XI~~ (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.2

Each MSCV must be inspected to ensure that it closes properly. This ensures that the safety analysis assumptions are met. The Frequency of this SR is based on Inservice Testing Program requirements and corresponds to the expected refueling cycle.

REFERENCES

1. FSAR, Section 10.2.

2. FSAR, Section 6.

3. FSAR, Section 14.

4. 10 CFR 50.67.

~~5. ASME, Boiler and Pressure Vessel Code, Section XI.~~
5. ASME code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

This SR verifies that the closure time of each MBFPDV(s), MFRV(s), and MFRV bypass valves is within required limits on an actual or simulated actuation signal. The closure times are assumed in the accident and containment analyses. The acceptance criteria for this SR do not include the 2 second delay associated with the ESFAS activation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves can not be tested at power because valve closure or even a part stroke exercise increases the risk of a valve closure and MBFP trip. This is consistent with the ASME Code~~,~~ ~~Section XI~~ (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program. The required Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the required Frequency.

REFERENCES

1. FSAR, Section 10.2.

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

2. ASME code for Operation and Maintenance of Nuclear Power Plants.

SURVEILLANCE REQUIREMENTS
(continued)

SR 3.7.5.1

maintained at a level that ensures a significant inventory is available as a heat sink before the AFW pump is required to refill the SG. These factors ensure that a significant amount of time would be available to complete any valve realignments needed to refill a SG when in Mode 4.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code~~, Section XI~~ (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test when SG pressure is < 600 psig.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage (i.e., unit at less than or equal to 97% power and in preparation for main generator breaker opening with no

(continued)

BASES

REFERENCES

1. FSAR, Section 10.2.

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

2. ASME code for Operation and Maintenance of Nuclear Power Plants.

3. Safety Evaluation Report (SER) for IP3 Amendment 225.