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August 21, 2008

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

Duke Energy Carolinas, LLC (Duke)
McGuire Nuclear Station Units 1 and 2
Docket Nos. 50-369 and 50-370

SUBJECT: License Amendment Request to Revise Technical Specification 5.5.8, "Inservice Testing Program," to Adopt Technical Specification Task Force (TSTF) Standard Technical Specification Change Travelers TSTF-479, Revision 0 and TSTF-497, Revision 0

Pursuant to 10 CFR 50.90, attached is a Duke Energy Carolinas, LLC (Duke) License Amendment Request (LAR) for McGuire Nuclear Station Facility Renewed Operating Licenses and Technical Specifications (TS). The proposed license amendment implements Technical Specification Task Force (TSTF) Change Travelers TSTF-479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a" and TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less". TSTF-479 and TSTF-497 revise the TS Administrative Controls section pertaining to requirements for the Inservice Testing Program, consistent 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 NRC issued letters on December 6, 2005, indicating the acceptability of TSTF-479, Revision 0 and October 4, 2006, indicating the acceptability of TSTF-497, Revision 0.

Implementation of this LAR in the Facility Operating Licenses and Technical Specifications will not impact the McGuire Updated Final Safety Analysis Report (UFSAR).

Duke Energy Carolinas requests approval of this LAR within one calendar year of the submittal date. Once approved, the amendment will be implemented within 60-days.

A047
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NRR

Enclosure 1 provides a description of the proposed change and the technical justification, an evaluation of significant hazards consideration pursuant to 10 CFR 50.92(c), and the following attachments.

Attachment 1 provides the existing TS pages marked-up to show the proposed change.

Attachment 2 provides the existing Bases pages marked-up to show the proposed change. These pages are being provided for information only.

Attachment 3 contains the retyped (clean) TS pages.

This submittal document contains no additional regulatory commitments.

In accordance with Duke's Administrative Procedures and Quality Assurance Program, this LAR has been reviewed and approved by the McGuire Plant Operations Review Committee and the Duke Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of this LAR is being sent to the designated official of the State of North Carolina.

If there are any questions or if additional information is needed, please contact K. L. Ashe at (704) 875-4535.

Sincerely,

Bruce Hamilton

Bruce H. Hamilton

Enclosure

cc with enclosure:

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Bruce H. Hamilton affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

Bruce Hamilton

Bruce H. Hamilton, Vice President, McGuire Nuclear Station

Subscribed and sworn to me: August 21, 2008
Date

Lori C. Bibby, Notary Public

My commission expires: July 1, 2012
Date

ENCLOSURE 1

Evaluation of the Proposed Change

Subject: License Amendment Request to Revise Technical Specification 5.5.8, "Inservice Testing Program," to Adopt Technical Specification Task Force (TSTF) Standard Technical Specification Change Travelers TSTF-479, Revision 0 and TSTF-497, Revision 0

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1. Technical Specification Page Markup
2. Bases Page Markups
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1. SUMMARY DESCRIPTION

This evaluation supports a request to amend Facility Operating Licenses NPF-9 and NPF-17 for McGuire Nuclear Station Units 1 and 2, respectively.

The proposed license amendment implements Technical Specification Task Force (TSTF) Change Travelers TSTF-479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a" and TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less". TSTF-479 and TSTF-497 revise the Improved Standard Technical Specifications (ISTS) Administrative Controls, TS 5.5.8, "Inservice Testing Program" consistent 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 NRC issued letters to the TSTF on December 6, 2005, indicating the acceptability of TSTF-479, Revision 0 and October 4, 2006, indicating the acceptability of TSTF-497. Duke Energy Carolinas LLC (Duke) is not proposing any variations or deviations from the TS changes described in TSTF-479, Revision 0 and TSTF-497, Revision 0.

2. DETAILED DESCRIPTION

McGuire Nuclear Station Units 1 and 2 TS 5.5.8 is being revised as shown below. The specific changes to Technical Specification 5.5.8 are shown in bold typeface.

Existing Requirement	Proposed Requirement
5.5.8 <u>Inservice Testing Program</u> 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 specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:	5.5.8 <u>Inservice Testing Program</u> 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 Operation and Maintenance of Nuclear Power Plants (ASME OM Code)** and applicable Addenda as follows:

Existing Requirement		**Proposed Requirement**	
ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities	ASME **OM** Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days	Weekly	At least once per 7 days
Monthly	At least once per 31 days	Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days	Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days	Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days	Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days	Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days	Biennially or every 2 years	At least once per 731 days
b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities; c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.		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** Code shall be construed to supersede the requirements of any TS.	

TS 5.5.8, "Inservice Testing Program," is being revised to delete references to Section XI of the ASME Boiler and Pressure Vessel Code and incorporate references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (i.e., the ASME OM Code). TS 5.5.8.b is also being revised to address the applicability of Surveillance Requirement (SR) 3.0.2 to other normal and accelerated Frequencies specified in the Inservice Testing (IST) Program.

Attachment 1 contains a marked-up version of the Technical Specifications showing the proposed changes. Attachment 3 provides a typed version of the Technical Specification pages. These typed Technical Specification pages are to be used for issuance of the proposed amendments.

Duke will make supporting changes to the Technical Specification Bases in accordance with TS 5.5.14, "Technical Specifications (TS) Bases Control Program." The affected TS Bases markup is included in Attachment 2; these pages are being submitted for information only.

3. TECHNICAL EVALUATION

Background

The ASME OM Code, provides rules for inservice testing of pumps and valves. The ASME OM Code replaced Section XI of the Boiler and Pressure Code for inservice testing of pumps and valves. The 1995 edition of the ASME OM Code is incorporated by reference into 10 CFR 50.55a. Since 10 CFR 50.55a(f)(4)(ii) requires that inservice testing comply with the requirements of the latest edition and addenda of the Code incorporated into 10 CFR 50.55a(b), TS 5.5.8 must be revised to reference the ASME OM Code.

Technical Analysis

Section XI of the ASME Code has been revised, on a continuing basis, to provide updated requirements for the inservice inspection and inservice testing of components. Until 1990, the ASME Code requirements addressing the inservice testing of pumps and valves were contained in Section XI. In 1990, the ASME published the initial edition of the ASME OM Code, which provides the rules for the inservice testing of pumps and valves. Since the establishment of the 1990 edition of the OM Code, the rules for inservice testing are no longer being updated in Section XI.

By Final Rule issued on September 22, 1999, the NRC amended 10 CFR 50.55a(f)(4)(ii) to require licensees to update their IST Program to the latest approved edition of the ASME OM Code incorporated by reference into 10 CFR 50.55a(b). TS 5.5.8 currently references the ASME Boiler and Pressure Vessel Code, Section XI, as the source for the IST Program requirements for ASME Code Class 1, Class 2, and Class 3 components.

Additionally, TS 5.5.8 is being revised to indicate the provisions of SR 3.0.2 are applicable to other inservice testing frequencies that are not specifically listed in the testing frequencies that are identified in TS 5.5.8. The IST Program may have frequencies for testing that are based on risk or other factors and do not conform to the standard testing frequencies specified in TS 5.5.8. The Frequency of the Surveillance may be determined through a mix of risk-informed and performance-based means in accordance with the IST Program. Application of SR 3.0.2 to other inservice testing frequencies is consistent with the guidance contained in NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants," January 2005, which indicated that the 25 percent extension of the interval specified in the frequency would apply to increased frequencies the same way that 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 requirements of the regulation take precedence over the TS.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

NRC regulation, 10 CFR 50.55a, defines the requirements for applying industry codes to each licensed nuclear power facility. The regulations require that during successive 120-month intervals, inservice inspection and inservice testing programs be developed using the latest edition and addenda of the ASME Code incorporated into paragraph (b) of 10 CFR 50.55a on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications identified in paragraph (b).

The proposed amendments do not:

(1) Alter the design or function of any system;
(2) Result in any changes in the qualifications of any component; and
(3) Result in the reclassification of any component's status in the areas of shared, safety-related, independent, redundant, and physically or electrically separated.

As such, there are no changes being proposed such that compliance with any of the regulatory requirements of 10 CFR 50.55a would come into question. Based on these 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) the issuance of the amendment will not be adverse to the common defense and security or to the health and safety of the public.

4.2 Precedents

1. Diablo Canyon Power Plant, Unit Nos. 1 and 2, "Issuance of Amendments Re: Administrative Changes to the Technical Specification 5.5.8, 'Inservice Testing Program' (TAC Nos. MD3975 and MD3976)," June 25, 2007, ADAMS Accession Numbers ML070990057 and ML070990074.

2. Watts Bar Nuclear Plant, Unit 1, "Issuance of Amendment Regarding the Inservice Testing Program (TAC No. MD2380) (TS-06-04)," December 18, 2006, ADAMS Accession Numbers ML063190441 and ML063550029.

3. Wolf Creek Generating Station, "Issuance of Amendment Re: Revision to Technical Specification 5.5.8 on the Inservice Testing Program (TAC No. MC9726)" November 15, 2006, ADAMS Accession Numbers ML062980233 and ML062980235.

4. Cooper Nuclear Station, "Issuance of Amendment Re: Technical Specification (TS) Changes Associated With Inservice Testing Program, Section 5.5.6, Under TS Programs and Manuals (TAC No. MD0335)," September 6, 2006, ADAMS Accession Numbers ML061440049 and ML062500301.

4.3 Significant Hazards Consideration

Duke Energy Carolinas, LLC (Duke) is submitting the proposed amendments to implement Technical Specification Task Force (TSTF) Change Traveler TSTF-479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a" and TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to frequencies of 2 years or Less". TSTF-479 and TSTF-497 revise the Improved Standard Technical Specifications (ISTS) Administrative Controls, TS 5.5.8, "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 NRC issued letters indicating the acceptability of the TSTFs on December 6, 2005 and October 4, 2006 respectively.

Duke has evaluated whether or not a significant hazards consideration is involved with the proposed amendment 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 TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. The proposed change incorporates revisions to the ASME Code as identified in the TSTFs referenced above.

   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. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released offsite and there is no increase in individual or cumulative occupational exposure.

   Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed change revises TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. The proposed change incorporates revisions to the ASME Code as identified in the TSTFs referenced above. The proposed change does not involve a modification to the physical configuration of the plant nor does it involve a 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 offsite and there is no increase in individual or cumulative occupational exposure.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   The proposed change revises TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. The proposed change does not involve a modification to the physical configuration of the plant nor does it change the methods governing normal plant operation. The proposed change incorporates revisions to the ASME Code as identified in the TSTFs referenced above.

   The safety function of the affected pumps and valves will be maintained.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Duke concludes that the proposed amendment does not involve a significant hazards consideration under the standard set forth in 10 CFR 50.92 (c), and accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusion

In conclusion, 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) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. Environmental Consideration

Duke has evaluated the proposed amendments for environmental considerations.

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, 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 effluents 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.

6. References

1. Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a."

2. ASME Code for Operation and Maintenance of Nuclear Power Plants, 1998 Edition through 2000 Addenda.

3. 10 CFR 50.55a, "Codes and standards"

4. NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants," January 2005.

5. Federal Register, Volume 64, No. 183, page 51370, "Final Rule, Industry Codes and Standards; Amended Requirements."

6. Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less."

ATTACHMENT 1

McGuire Units 1 and 2 Technical Specifications

Page Markup

5.5 Programs and Manuals (continued)

5.5.8 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:

applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code)

a. Testing frequencies ~~specified in Section XI of the ASME Boiler and Pressure Vessel Code~~ and applicable Addenda as follows:

OM

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

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

and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program

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

OM

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

5.5.9 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

(continued)

ATTACHMENT 2

McGuire Units 1 and 2 Technical Bases

Page Markups

(Provided for information only)

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 ≤ 300°F within 12 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 300°F, 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.

OM

The pressurizer safety valve setpoint is + 3% and - 2% of the nominal setpoint of 2485 psig 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. UFSAR, Chapter 15.

3. UFSAR Section 5.2.

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

ASME Code for Operation and Maintenance of Nuclear Power Plants.

5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

ACTIONS (continued)

H.1 and H.2

If the Required Actions of Condition F or G are not met, then 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 6 hours and to MODE 4 within 12 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. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the Frequency of 92 days is the ASME OM Code, ~~Section XI~~ (Ref. 4). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, 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 otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, 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 (i.e., completion of the Required Actions fulfills the SR).

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 18 months is based on a typical refueling cycle and industry accepted practice.

The SR is modified by a Note which states that the SR is required to be performed in MODE 3 or 4 when the temperature of the RCS cold legs is > 300°F consistent with Generic Letter 90-06 (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.3

The Surveillance demonstrates that the emergency nitrogen supply can be provided and is performed by transferring power from normal air supply to emergency nitrogen supply and cycling the valves. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. UFSAR, Section 15.4.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

   ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," Pursuant to 10 CFR 50.54(f) (Generic Letter 90-06).

SURVEILLANCE REQUIREMENTS (continued)

accordance with the Inservice Testing Program. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet the Required Actions of this LCO.

The RHR suction valves are verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valves remain open.

OM

The ASME Code, ~~Section XI~~ (Ref. 9), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.4

The RCS vent of ≥ 2.75 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that cannot be locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. *A removed pressurizer safety valve fits this category.*

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

SR 3.4.12.5

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.6

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ 300°F and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the allowed maximum limits. PORV actuation could depressurize the RCS and is not required.

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during this time.

A Note has been added indicating that this SR is required to be met 12 hours after decreasing RCS cold leg temperature to ≤ 300°F. The test must be performed within 12 hours after entering the LTOP MODES.

SR 3.4.12.7

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. UFSAR, Section 5.2.
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
8. Generic Letter 90-06.
9. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

ASME Code for Operation and Maintenance of Nuclear Power Plants.

10. Duke letter to NRC, "Cold Leg Accumulator Isolation Valves", dated September 8, 1987.

ACTIONS (continued)

B.1 and B.2

If leakage cannot be reduced, or the other Required Actions accomplished, 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 MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

The RHR interlock prevents the RHR suction isolation valves inadvertent opening at RCS pressures in excess of the RHR systems design pressure. If the RHR interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the interlock function.

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 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 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 9) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, ~~Section XI~~

OM

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. UFSAR Table 5-50.
7. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
8. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

   ASME Code for Operation and Maintenance of Nuclear Power Plants.
9. 10 CFR 50.55a(g).

SURVEILLANCE REQUIREMENTS (continued)

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.

This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

ECCS piping is verified to be water-filled by venting to remove gas from accessible locations susceptible to gas accumulation. Alternative means may be used to verify water-filled conditions (e.g., ultrasonic testing or high point sightglass observation). Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

OM

SR 3.5.2.4

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 the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

OM

ACTIONS (continued)

B.1 and B.2

If the affected containment spray train 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 6 hours and to MODE 5 within 84 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. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or computer status indication, that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

OM

SR 3.6.6.2

Verifying that 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 head 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 bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice 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.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. UFSAR, Section 6.2.
3. 10 CFR 50.49.
4. 10 CFR 50, Appendix K.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

ASME Code for Operation and Maintenance of Nuclear Power Plants.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

With one or more MSSVs inoperable, reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. For example, if one MSSV is inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 20%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced. This is accomplished by reducing THERMAL POWER by the necessary amount to conservatively limit the energy transfer to all steam generators, consistent with the relief capacity of the most limiting steam generator.

The maximum power level specified for the power range neutron flux high trip setpoint with inoperable MSSVs must ensure that power is limited to less than the heat removal capacity of the remaining OPERABLE MSSVs. The reduced high flux trip setpoint also ensures that the reactor trip occurs early enough in the loss of load/turbine trip event to limit primary to secondary heat transfer and preclude overpressurization of the primary and secondary systems. To calculate this power level, the governing equation is the relationship $q = m\,\Delta h$, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure (assuming no subcooled feedwater). The algorithm use is consistent with the recommendations of the Westinghouse Nuclear Safety Advisory Letter, NSAL-94-001, dated January 20, 1994 (Ref. 7). Additionally, the calculated values are reduced by 9% to account for instrument and channel uncertainties.

6

The allowed Completion Time of 4 hours provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time and provides sufficient time to reduce the trip setpoints. The adjustment of the trip setpoints is a sensitive operation that may inadvertently trip the Reactor Protection System.

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 OM Code, ~~Section XI~~ (Ref. 5), requires that safety and relief valve tests be performed ~~in accordance with ANSI/ASME OM-1-1987 (Ref. 6)~~. According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

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.

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 10.3.1.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. UFSAR, Section 15.2.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

   ASME Code for Operation and Maintenance of Nuclear Power Plants.

~~6. ANSI/ASME OM-1-1987.~~

6

~~7.~~ Westinghouse Nuclear Safety Advisory Letter, NSAL-94-001, Dated January 20, 1994.

ACTIONS (contd)

D.1 and D.2

If the MSIVs 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 ≤ 8.0 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 should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME OM Code, ~~Section XI~~ (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

Testing shall be performed with both spring force and the motive force provided by Instrument Air (VI) simultaneously. Leak-rate testing of the MSIV air control system shall be performed prior to returning the unit to operation following a refueling outage.

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5 exercising requirements. 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.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 6.2.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. 10 CFR 100.11.
5. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ ASME Code for Operation and Maintenance of Nuclear Power Plants.

ACTIONS (continued)

D.1

With two inoperable valves in the same flow path, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFCV, or otherwise isolate the affected flow path.

E.1 and E.2

If the MFIV(s), MFCV(s), MFCV's bypass valve(s), and MFW/AFW NBV(s) cannot be restored to OPERABLE status, or closed, or isolated 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 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 3.7.3.1

This SR verifies that the closure time of each MFIV, MFCV, MFCV's bypass valve, and MFW/AFW NBV is ≤ 10 seconds on an actual or simulated actuation signal. The MFIV and MFCV closure times are assumed in the accident and containment analyses. This Surveillance is 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 OM Code, ~~Section XI~~ (Ref. 3), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program.

BASES

REFERENCES

1. UFSAR, Section 10.4.7.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

ASME Code for Operation and Maintenance of Nuclear Power Plants.

SURVEILLANCE REQUIREMENTS (continued)

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

OM

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 3). 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. 3) (only required at 3 month intervals) satisfies this requirement.

OM

The Frequency for this SR is in accordance with the Inservice Testing Program.

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. The test should be conducted within 24 hours of the steam pressure exceeding 900 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 18 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. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train may already be aligned and operating.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 ESFAS 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 may already be operating and the autostart function is not required. The 18 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 can be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. The test should be conducted within 24 hours of the steam pressure exceeding 900 psig. Note 2 states that the SR is not required in MODE 4. In MODE 4, the required pump may already be operating and the autostart function is not required. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump if it were not in operation.

REFERENCES

1. UFSAR, Section 10.4.7.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

ASME Code for Operation and Maintenance of Nuclear Power Plants.

ATTACHMENT 3

Retyped McGuire Units 1 and 2 Technical Specifications Page

Remove	Insert
5.5-6	5.5-6

5.5.8 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 Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM 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 Code shall be construed to supersede the requirements of any TS.

5.5.9 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

(continued)