Entergy

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Pete Dietrich
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JAFP-09-0131
November 23, 2009

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

SUBJECT: Application for Amendment to Modify Technical Specifications Section 5.5.7, Inservice Testing Program, to Reference the Current Code of Record
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-59

REFERENCE: Entergy Letter, Proposed Relief Requests for the James A. FitzPatrick Nuclear Power Plant Fourth Interval In-Service Testing Program, JAFP-07-0050, dated April 11, 2007

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (Entergy) hereby requests an amendment to the Technical Specifications (TS) for the James A. FitzPatrick Nuclear Power Plant (JAF).

This license amendment submittal requests replacing the TS 5.5.7 (Inservice Testing Program) references from the ASME Boiler and Pressure Vessel Code to the current code of record, the ASME Operation and Maintenance Nuclear Power Plants Code. This is an administrative amendment, to maintain the TS current with the NRC accepted code of record for JAF.

Attachment 1 provides a description and evaluation of the proposed TS additions.
Attachment 2 provides the proposed TS changes as marked up pages.
Attachment 3 provides the proposed TS changes in final typed format with change bars.
Attachment 4 provides the proposed TS Bases changes as marked up pages.

The Bases changes are provided for NRC information only. The final TS Bases pages will be submitted with a future update consistent with the code change in TS 5.5.7.

Entergy requests NRC approval of the proposed TS amendment by May 23, 2010, with the amendment being implemented within 30 days from approval.

In accordance with 10 CFR 50.91, a copy of this application, with the associated attachments, is being provided to the designated New York State official.

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NRR

There is no new commitment made in this letter.

Questions concerning this request may be addressed to Mr. Joseph Pechacek, Licensing Manager, at (315) 349-6766.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 23rd day of November 2009.

Sincerely,

Pete Dietrich

PD/JP/ed

Attachments:
1. Application for Amendment to Modify Technical Specifications Section 5.5.7, Inservice Testing Program, to Reference the Current Code of Record
2. Proposed Technical Specification Changes (Marked up)
3. Proposed Technical Specification Changes (Final Typed)
4. Proposed Technical Specification Bases Changes (Marked up) (Information Only)

cc: next page

cc:

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Attachment 1

Description and Evaluation

Application for Amendment to Modify Technical Specifications Section 5.5.7, Inservice Testing Program, to Reference the Current Code of Record
(3 Pages)

1.0 DESCRIPTION

The proposed amendment would modify the Technical Specification (TS) 5.5.7, Inservice Testing Program, by replacing the references from the ASME Boiler and Pressure Vessel Code to the current code of record, the ASME Operation and Maintenance Nuclear Power Plants Code (ASME OM Code). This is an administrative amendment to maintain the TS current with the NRC accepted code of record for JAF.

2.0 PROPOSED CHANGES

Currently, TS 5.5.7, Item a. refers to "Section XI of the ASME Boiler and Pressure Vessel Code" and to the "ASME Boiler and Pressure Vessel Code." However, those references are outdated and should be changed to the ASME OM Code, so that the TS is current with respect to the code of record for the James A. FitzPatrick Nuclear Power Plant (JAF) Inservice Testing Program as referenced in the Fourth Inservice Testing Interval Relief Requests submitted to the NRC on April 11, 2007 (Reference 1).

TS Bases will be revised to also address the ASME OM Code.

3.0 BACKGROUND

10 CFR 50.55a, Codes and standards, (Reference 2) has been modified, to allow the use of the ASME OM Code for inservice testing, affectively replacing Section XI of the ASME Boiler and Pressure Vessel Code. Plus, Regulatory Guide 1.192 (Reference 3) was issued to provide guidance on the acceptability of the ASME OM Code.

Consistent with References 1, 2, and 3, the JAF code of record for its Inservice Testing Program has been changed to the ASME OM Code. This is reflected in the Reference 4 NRC Safety Evaluation which states "The James A. FitzPatrick Nuclear Power Plant fourth 10-year IST interval began on September 27, 2007. The program was developed in accordance with the 2001 Edition through 2003 Addenda of the ASME OM Code."

Currently, JAF TS 5.5.7 still references to the ASME Boiler and Pressure Vessel Code, and thus, those references need to be revised.

4.0 TECHNICAL ANALYSIS

There is no technical change being requested. TS changes simply revise two outdated references to reflect the NRC accepted code of reference for the JAF Inservice Testing Program Fourth 10-Year Interval.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.
The proposed TS changes are non-technical, and are provided for consistency. There is no plant change involved, and thus, proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed TS changes are non-technical, i.e., there is no plant change involved, and thus, do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed TS changes are non-technical, i.e., there is no plant change involved. The changes are consistent with the regulations, and only update the TS to refer to the current code of reference. No design or safety margin is involved. Therefore, the proposed changes do not involve a significant reduction in any margin of safety.

5.2 Applicable Regulatory Requirements / Criteria

10 CFR 50.55a allows the use of the ASME OM Code for inservice testing. Consistent with 10 CFR 50.55a, the JAF code of record for its Fourth 10 Year Interval Inservice Testing Program has been changed to the ASME OM Code, and the TS are to be revised to reflect the change to the ASME OM Code.

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 approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL ASSESSMENT

A review has determined that the proposed changes 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 changes do 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 changes meet 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 changes.

7.0 REFERENCE(S)

1. Entergy Letter, Proposed Relief Requests for the James A. FitzPatrick Nuclear Power Plant Fourth Interval In-Service Testing Program, JAFP-07-0050, dated April 11, 2007
2. USNRC, Code of Federal Regulations, Section 10 CFR 50.55a, "Codes and standards."
3. USNRC, Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," Revision 0, June 2003.
4. NRC (Mark G. Kowal) letter to Entergy (Michael A. Balduzzi), "JAMES A. FITZPATRICK NUCLEAR POWER PLANT - RELIEF REQUESTS FOR THE FOURTH INTERVAL INSERVICE TESTING PROGRAM (TAC NOS. MD5396, MD5397, MD5398, MD5399, MD5400, MD5401, MD5402, MD5403, AND MD5404)," November 27, 2007.

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Attachment 2

Proposed Technical Specification Changes (Marked up)

<u>Page</u>

5.5-6

5.5 Programs and Manuals

5.5.6 Primary Containment Leakage Rate Testing Program (continued)

d. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.7 Plant Staff

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

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

ASME ~~Boiler and Pressure Vessel~~ 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 for performing inservice testing activities;

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

(continued)

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Attachment 3

Proposed Technical Specification Changes
(Final Typed)

5.5 Programs and Manuals

5.5.6 Primary Containment Leakage Rate Testing Program (continued)

d. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.7 Plant Staff

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

a. Testing Frequencies specified in the ASME OM Code and applicable Addenda are as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Quarterly or every 3 months	At least once per 92 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

(continued)

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Attachment 4

Proposed Technical Specification Bases Changes (Marked up)
(Information Only)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.6

Cycling the recirculation pump discharge valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection signal, these valves are required to close to ensure full LPCI subsystem flow injection in the reactor via the recirculation jet pumps. De-energizing the valve in the closed position will also ensure the proper flow path for the LPCI subsystem. Acceptable methods of de-energizing the valve include de-energizing breaker control power, racking out the breaker or removing the breaker.

The specified Frequency is once during reactor startup before THERMAL POWER is > 25% RTP. However, this SR is modified by a Note that states the Surveillance is only required to be performed if the last performance was more than 31 days ago. Verification during reactor startup prior to reaching > 25% RTP is an exception to the normal Inservice Testing Program generic valve cycling Frequency of 92 days, but is considered acceptable due to the demonstrated reliability of these valves. If the valve is inoperable and in the open position, the associated LPCI subsystem must be declared inoperable.

SR 3.5.1.7, SR 3.5.1.8, and SR 3.5.1.9

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME Operation and Maintenance Nuclear Power Plants Code (ASME OM Code, Ref. 14)~~, Section XI~~, requirements for the ECCS pumps) to verify that the ECCS pumps will develop at least the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 10. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values may be established during preoperational testing.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.1.13 (continued)

The Frequency of 24 months on a STAGGERED TEST BASIS ensures that both solenoids for each ADS valve are alternately tested. The Frequency is based on the need to perform the Surveillance under the conditions that apply during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.4.3.
2. UFSAR, Section 6.4.4.
3. UFSAR, Section 6.4.1.
4. UFSAR, Section 6.4.2.
5. NEDC-31317P, Revision 2, James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis, April 1993.
6. UFSAR, Section 14.6.1.5.
7. UFSAR, Section 14.6.1.3.
8. 10 CFR 50, Appendix K.
9. UFSAR, Section 6.5.
10. 10 CFR 50.46.
11. 10 CFR 50.36(c)(2)(ii).
12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), Recommended Interim Revisions to LCOs for ECCS Components, December 1, 1975.
13. UFSAR, Section 4.4.5.
14. ASME Operation and Maintenance Nuclear Power Plants Code, 2001 Edition through 2003 Addenda.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.9.1 (continued)

also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR Containment Spray System is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

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

SR 3.6.1.9.2

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

SR 3.6.1.9.3

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

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Table 5.2-1.

2. UFSAR, Section 5.2.4.4.

3. UFSAR, Section 14.6.

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

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

6. ASME~~,~~ Operation and Maintenance Nuclear Power Plants Code, 2001 Edition through 2003 Addenda.~~Boiler and Pressure Vessel Code, Section XI.~~

7. JAF-RPT-06-00063, Rev. 0, Drywell Spray Flow Rate Design and Licensing Bases, September 19, 2006.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1 (continued)

event requiring initiation of the system is low, and the system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2

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

REFERENCES

1. UFSAR, Section 14.6.1.3.3.
2. GE-NE-T23-00737-01, James A. FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, August 1996.
3. NEDC-24361-P, James. A FitzPatrick Nuclear Power Plant Suppression Pool Temperature Response, August 1981.
4. 10 CFR 50.36(c)(2)(ii).
5. ASME Operation and Maintenance Nuclear Power Plants Code, 2001 Edition through 2003 Addenda.~~ASME, Boiler and Pressure Vessel Code, Section XI.~~