



Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
James A. Fitzpatrick NPP
P.O. Box 110
Lycoming, NY 13093

Joseph Pechacek
Licensing Manager

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U.S. Nuclear Regulatory Commission
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Subject: Response to Request For Additional Information Regarding Proposed Relief Request VRR-06 for the James A. FitzPatrick Nuclear Power Plant Fourth Interval In-Service Testing Program (TAC No. ME1818).
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-59

- References:
1. Entergy Letter, JAFP-09-0088, Proposed Relief Request VRR-06 for the James A. FitzPatrick Nuclear Power Plant Fourth Interval In-Service Testing Program (TAC No. ME1818), dated July 31, 2009
 2. NRC Request For Additional Information Regarding Proposed Relief Request VRR-06 for the James A. FitzPatrick Nuclear Power Plant Fourth Interval In-Service Testing Program (TAC No. ME1818), dated August 21, 2009

Dear Sir or Madam:

On July 31, 2009 Entergy Nuclear submitted a proposed relief request for the James A. FitzPatrick Nuclear Power Plant (JAF) Fourth Interval In-Service Testing Program (Reference 1). On August 21, 2009 JAF received two Request for Additional Information (RAI) questions from the NRC Staff regarding the submittal (Reference 2). On August 24, 2009 JAF and the NRC Staff participated in a conference call to clarify these questions. As a result of those clarifications, JAF is submitting this revision to Request for Relief VRR-06.

In accordance with 10 CFR 50.55a(a)(3)(i), this revised request proposes to implement an alternative to the 60-month test interval required by ASME OM Code, 2001 Edition through 2003 Addenda, Appendix I, Paragraph I-1320(a). The proposed alternative is a 72-month test interval for Class 1 pressure relief valves with a minimum of 20% of the SRV group being tested within any 24-month interval. This 20% shall consist of valves that have not been tested during the current 72-month interval. The test interval for any individual valve that is in service shall not exceed 72 months except that a six month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods. The proposed alternative test interval accommodates the JAF 24-month fuel cycle and continues to provide an acceptable level of quality and safety.

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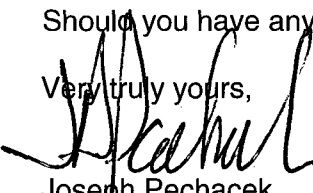
In accordance with 10 CFR 50.55(f)(4)(iv), this revised request also proposes to use a portion of a subsequent edition of the ASME OM Code. Specifically, ASME OM Code 2004 Edition no addenda, Appendix I, Paragraph I-3410(d).

The fourth interval began October 1, 2007 and this relief will be applicable for the duration of the fourth interval. JAF requests approval of the enclosed revised relief request by December 15, 2009 to support submittal of an application for amendment to the JAF Technical Specification Surveillance Requirements to reflect this relief.

There are no commitments made in this letter.

Should you have any questions, please contact Mr. Joseph Pechacek at (315) 349-6766.

Very truly yours,



Joseph Pechacek
Licensing Manager

JP:mh

Enclosure 1: RAI Questions and Responses

Enclosure 2: Revised James A. FitzPatrick Nuclear Power Plant Fourth Interval
In-Service Testing Program Relief Request VRR-06

cc: Mr. Samuel Collins
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406-1415

Mr. Bhalchandra Vaidya, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-8-C2A
Washington, DC 20555-0001

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
James A. FitzPatrick Nuclear Power Plant
P.O. Box 136
Lycoming, New York 13093

JAFP-09-0102

Enclosure 1

RAI QUESTIONS and RESPONSES

ENCLOSURE 1
RAI QUESTIONS and RESPONSES

RAI QUESTION 1:

Relief Request VRR-06 states that approximately one third of the main steam safety relief valve main assemblies (4, 4, and 3 main assemblies over a three-cycle period) will be sent to an offsite vendor and tested with steam pressure each refueling outage. Please describe how each main assembly will be tested with steam pressure.

RESPONSE:

In the "Current Testing at Outside Facilities" section of the revised Relief Request VRR-06 (Enclosure 2 of this submittal) JAF states, "Post maintenance testing includes initial valve leakage testing, safety mode valve actuation to satisfy requirements for set pressure, reseal pressure, main disc stroke time, and final leakage testing for both the pilot and main body using saturated steam. Final seat leakage tests are performed at approximately 1070 psig. Upon successful test completion, each valve receives written certification from the vendor and is returned to JAF for reinstallation. To receive certification, the valve must have zero seat leakage and meet the acceptance criteria of +/- 1% for set pressure using saturated steam. These tests meet the requirements of ASME OM Code 2001 Edition through 2003 Addenda, Appendix I, Paragraph I-3310 and Technical Specifications SR 3.4.3.1."

RAI QUESTION 2:

Relief Request VRR-06 also proposes to setpoint test the main steam safety relief valves every six years in accordance with Code Case OMN-17, "Alternative Rules for Testing ASME Class 1 Pressure Relief/Safety Valves." Code Case OMN-17 cannot be approved by the Nuclear Regulatory Commission until it is published by the ASME. Please resubmit Request VRR-06 without referencing Code Case OMN-17. The submittal should address all the provisions in Code Case OMN-17. The submittal should also address why it is a hardship or unusual difficulty without a compensating increase in the level of quality and safety to test the main steam safety relief valves every 5 years in accordance with I-1320(a) of Mandatory Appendix I of the ASME OM Code.

RESPONSE:

As clarified in the conference call with the NRC Staff on August 24, 2009 JAF is proposing an alternative to the test interval required by Paragraph I-1320(a) of Mandatory Appendix I of the ASME OM Code 2001 Edition through 2003 Addenda in accordance with 10 CFR 55a(a)(3)(i). The proposed alternative test interval accommodates the JAF 24 month fuel cycle and continues to provide an acceptable level of quality and safety. Since the alternative is being proposed in accordance with 10 CFR 55a(a)(3)(i) rather than 10 CFR 55a(a)(3)(ii) hardship and unusual difficulty do need to be addressed.

The attached revision to VRR-06 incorporates the JAF Response to these requests for additional information and supersedes the original Relief Request in its entirety.

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Enclosure 2

**James A. FitzPatrick Nuclear Power Plant
Fourth Interval In-Service Testing Program
Relief Request VRR-06 Revision 1**

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

10 CFR 50.55a Request VRR-06 Rev.1

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i), on the basis that the proposed alternative provides an acceptable level of quality and safety
and

Proposed Subsequent Code Edition and Addenda (or Portion) in Accordance with 10 CFR 50.55a(f)(4)(iv)

System:

NUCLEAR BOILER AUTOMATIC DEPRESSURIZATION SYSTEM

ASME Code Components Affected:

02RV-71A, B, C, D, E, F, G, H, J, K, L

Component /System Function:

Eleven Safety/Relief Valves (SRVs) are installed on the main steam lines between the reactor vessel and the inboard main steam isolation valves. Each SRV discharges via a separate tailpipe to a point below the water level in the suppression pool. SRVs open:

- In the safety mode on high reactor pressure, to provide primary overpressure protection to the reactor coolant pressure boundary.
- In the relief mode when actuated by the SRV Electric Lift logic on high reactor pressure, as a backup to the safety mode actuation.
- In the relief mode when manually actuated by individual control switches in the Control Room, or by individual control switches in the Remote Shutdown system.
- For seven of the eleven SRVs, in the relief mode when actuated by the Automatic Depressurization System (ADS) logic of the Emergency Core Cooling Systems (ECCS). The ADS function is to rapidly reduce reactor pressure to within the capacity of low pressure ECCS pumps in the event of a small or intermediate break Loss of Coolant Accident with the High Pressure Coolant Injection System (HPCI) unable to maintain level due to equipment failure or break size.

Applicable Code Edition and Addenda:

ASME OM CODE 2001 Edition to 2003 Addenda

ASME OM CODE 2004 Edition no Addenda

Applicable Code Requirements:

ASME OM CODE 2001 Edition to 2003 Addenda, Appendix I, paragraph I-1320(a), "Test Frequencies, Class 1 Pressure Relief Valves" requires that Class 1 pressure relief valves shall be tested at least once every 5 years, starting with initial electric power generation. No maximum limit is specified for the number of valves to be tested within each interval; however, a minimum of 20% of the valves from each valve group shall be tested within any 24-month interval. This 20% shall consist of valves that have not been tested during the current 5-year interval, if they exist. The test interval for any individual valve shall not exceed 5 years.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

10 CFR 50.55a Request VRR-06 Rev.1

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i), on the basis that the proposed alternative provides an acceptable level of quality and safety
and

Proposed Subsequent Code Edition and Addenda (or Portion) in Accordance with 10 CFR 50.55a(f)(4)(iv)

Applicable Code Requirements(continued):

ASME OM CODE 2004 Edition no Addenda, Appendix I, Paragraph I-3410 (d) "Each valve with an auxiliary actuating device that has been removed for maintenance or testing and reinstalled after meeting the requirements of I-3310, shall have the electrical and pneumatic connections verified either through mechanical / electrical inspections or test prior to the resumption of electrical power generation. Main disc movement and set pressure verification are not required.

Reason For Request:

This 4th Interval request is based on Appendix I of the ASME OM Code 2001 Edition to 2003 Addenda and Appendix I of the ASME OM Code 2004 Edition. This request proposes an alternative interval for Testing ASME Class 1 Pressure Relief/Safety Valves. As discussed under "Applicable Code Requirements Appendix I, Paragraph I-1320(a) includes a requirement to test 100% of the SRV population every 5 years. JAF proposes an alternate interval. Safety valves shall be tested at least once every 72 months (6 years) with a minimum of 20% of the SRV group being tested within any 24-month interval. This 20% shall consist of valves that have not been tested during the current 72-month interval. The test interval for any individual valve that is in service shall not exceed 72 months except that a 6-month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods. This proposed alternative is based on aligning the test interval with JAF's 24 month fuel cycle.

Exercising of the SRV after reinstallation as required by Appendix I, Paragraph I-3410 of the ASME OM Code 2001 Edition to 2003 Addenda, can only be performed during reactor startup when there is sufficient steam pressure to actuate the main disk. This test requirement was modified by Appendix I, Paragraph I-3410(d) of the ASME OM Code 2004 Edition to require only electrical and pneumatic connection verifications either through mechanical / electrical inspections or test prior to the resumption of electrical power generation; main disc movement and set pressure verification are not required, after satisfying the requirements of Paragraph I-3310.

System Description:

There are a total of 11 Safety/Relief Valves (SRVs) installed on the main steam lines at JAF. The SRVs are dual-function valves capable of being independently opened in either the safety or relief mode of operation. In the safety mode, the SRVs are actuated directly by main steam pressure. In the relief mode, each SRV can be actuated by either of two SOVs per valve that supply pneumatic pressure to the valve actuator. One SOV can be actuated by a manual control switch in the Control Room, by the Electric Lift logic, or (for 7 of the 11 SRVs) by the ADS logic. The other SOV can be actuated by a manual control switch at the remote shutdown panel in the Reactor Building.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

10 CFR 50.55a Request VRR-06 Rev.1

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i), on the basis that the proposed alternative provides an acceptable level of quality and safety
and

Proposed Subsequent Code Edition and Addenda (or Portion) in Accordance with 10 CFR 50.55a(f)(4)(iv)

System Description (continued):

The current SRVs are Target Rock model 7567F two-stage SRVs. In the safety mode, the spring loaded pilot stage open when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. The pilot valve ports steam from the main stage piston chamber to the tailpipe, creating a differential pressure across the main stage piston that opens the main stage. In the relief mode, the pneumatic actuator lifts the pilot preload spring off of the pilot disc; steam pressure then opens the pilot and main stage as in the safety mode.

A modification is currently scheduled to be installed beginning in October 2010 to replace the current model 7567F two-stage SRVs with Target Rock model 0867F three-stage SRVs, which have better leakage performance and are less susceptible to setpoint changes caused by corrosion bonding. Three SRVs are scheduled to be replaced in 2010, four in 2012, and four in 2014. The three-stage SRVs will be actuated by the same methods as the two-stage valves. In the safety mode, steam from the valve inlet expands a metal bellows; at the setpoint, bellows expansion is sufficient to lift the pilot valve. The pilot valve discharges to the second stage piston chamber, opening the reverse seated second stage disc against steam pressure and spring force. The second stage valve ports steam from the main stage piston chamber to the tailpipe, creating a differential pressure across the main stage piston that opens the main stage. In the relief mode, the pneumatic actuator opens the second stage directly, which opens the main stage as in the safety mode; the pilot stage is not involved in pneumatic operation.

Current Testing at JAF:

Testing of JAF SRVs is performed to satisfy Technical Specifications Surveillance Requirements (TSSRs) and the ASME OM Code 2001 Edition through 2003 Addenda, "Code for Operation and Maintenance of Nuclear Power Plants with addenda." Certain tests are performed with the SRVs installed (in situ), while others are performed as "bench tests" after the valve is removed and transported to a qualified maintenance and testing facility. Current requirements are as follows:

1. TSSRs 3.4.3.2 and 3.5.1.13 verify each SRV opens when manually actuated.
2. Remote manual actuation is also required by ASME OM Code 2001 Edition through 2003 Addenda, Appendix I, paragraph I-3410(d), to verify open and close capability of the valve before resumption of electric power generation. This applies to valves that have been either maintained in place, or removed for maintenance and testing and reinstalled.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

10 CFR 50.55a Request VRR-06 Rev.1

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i), on the basis that the proposed alternative provides an acceptable level of quality and safety
and

Proposed Subsequent Code Edition and Addenda (or Portion) in Accordance with 10 CFR 50.55a(f)(4)(iv)

Current Testing at Outside Facilities:

During each refueling outage, the 11 SRV pilot assemblies are removed and shipped to an offsite qualified maintenance and testing facility. At least one main body is also removed each cycle for disassembly and inspection in accordance with UFSAR Section 4.4.7. Pilots receive "as-found" testing, which includes visual inspection, leakage testing, and as-found set pressure testing. The tests are performed on the pilots prior to maintenance on the valve using saturated steam. Main bodies are visually inspected. The pre-test leakage on the pilot is measured at 1050 psig using steam meeting the requirements of ASME OM Code 2001 Edition Through 2003 Addenda, Appendix I, Paragraph I-3310 (a) (b) & (c).

Following the "as-found" testing of the SRV Pilots, the SRV pilots and main bodies are given a dimensional inspection followed by refurbishment, if required. This work is performed by the qualified valve supplier.

Post maintenance testing includes initial valve leakage testing; safety mode valve actuation to satisfy requirements for set pressure, reseal pressure, main disc stroke time; and final leakage testing for both the pilot and main body using saturated steam. Final seat leakage tests are performed at approximately 1070 psig. Upon successful test completion, each valve receives written certification from the vendor and is returned to JAF for reinstallation. To receive certification, the valve must have zero seat leakage and meet the acceptance criteria of +/- 1% for set pressure using saturated steam. These tests meet the requirements of ASME OM Code 2001 Edition through 2003 Addenda, Appendix I, Paragraph I-3310 and Technical Specifications SR 3.4.3.1.

General Change Justification:

Leaking SRVs create operational problems associated with the suppression pool such as increases in both pool temperature and level, which, in turn, requires more frequent use of the suppression pool cooling mode of the Residual Heat Removal (RHR) system.

As described previously, each SRV pilot assembly and at least one of the main bodies are removed during each refuel outage and are tested at a qualified maintenance and testing facility. All eleven main bodies have been tested over the past three cycles to comply with SIL-646. The as found testing is performed within 12 months of removal, meeting the OM code requirements. The valves are refurbished the following year as necessary, to meet the acceptance criteria of zero leakage just prior to the refuel outage, and are certified in writing as being leak free. The valves are then returned to JAF for reinstallation in the plant and proper pilot operation is confirmed through leak rate testing of the pneumatic accumulator and piping up to the SOVs followed by manual lift at reactor power.

Several aspects of SRV design and operation can contribute to valve leakage. As mentioned earlier, these include test pressure, pilot valve disc and rod configuration, and system and valve cleanliness. Actuation of the SRVs after laboratory testing by any means allows these contributors to impact the ability of the valve to re-close completely. JAF has made significant efforts to minimize the effects of these contributors.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

10 CFR 50.55a Request VRR-06 Rev.1

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i), on the basis that the proposed alternative provides an acceptable level of quality and safety
and

Proposed Subsequent Code Edition and Addenda (or Portion) in Accordance with 10 CFR 50.55a(f)(4)(iv)

General Change Justification (continued):

Additionally, reducing challenges to the SRVs is a recommendation of NUREG-0737; "TMI Action Plan Requirements" item II.K.3. (16). This recommendation is based on a stuck open SRV being a possible cause of a Loss of Coolant Accident (LOCA). The proposed use of a portion of a later approved code edition is consistent with this recommendation. By modifying the in-situ testing requirements in Paragraph I-3410(d) for refurbished valves that meet the testing requirements of Paragraph I-3310, unnecessary cycling of the valves is avoided.

JAF currently uses ASME OM Code 2001 Edition through 2003 Addenda, Appendix I, Paragraph I-1320, "Test Frequencies, Class I Pressure Relief Valves." This establishes a 60-month interval for SRV testing. JAF proposes to use an alternative 72-month interval for testing ASME Class 1 Pressure Relief/Safety Valves. The proposed alternative 72-month interval aligns the testing interval with the 24-month refueling cycle at JAF. The revised interval accommodates the JAF 24-month fuel cycle and continues to provide an acceptable level of quality and safety.

Proposed Alternative In Accordance With 10 CFR 50.55a(a)(3)(i):

JAF proposes to implement a 72-month test interval for Class 1 pressure relief valves with a minimum of 20% of the SRV group being tested within any 24-month interval. This 20% shall consist of valves that have not been tested during the current 72-month interval. The test interval for any individual valve that is in service shall not exceed 72 months except that a six month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods.

Each refueling outage, 100% of the pilot assemblies and 27%-36% of the main bodies (4, 4, and 3 main assemblies over a three cycle period) will be sent to an qualified maintenance and testing facility and tested with steam pressure. As a result, even though actual valve movement is not performed after the SRV is re-installed in the plant, all pilot assemblies are tested with steam pressure once per cycle and all the main bodies are tested with steam pressure at least once every three cycles. This same process will be followed for the three-stage valves.

For two-stage SRVs, following installation, the electrical connections are verified by continuity testing from the respective control panels. The pneumatic connections to the SOV are verified by accumulator leak testing. Pneumatic connections downstream of the SOV cannot be tested without actuating the SRV pilot; these connections are verified by inspections during installation in accordance with approved maintenance procedures.

For three-stage SRVs, following installation, the electrical and pneumatic connections will be verified by energizing the SOVs using the respective control switches and inspecting the pneumatic actuator for movement and leakage. This test will actuate the SRV second stage, but operating experience at other plants indicates that it does not initiate second stage leakage or otherwise damage the valve when performed with no steam pressure.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

10 CFR 50.55a Request VRR-06 Rev.1

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i), on the basis that the proposed alternative provides an acceptable level of quality and safety
and

Proposed Subsequent Code Edition and Addenda (or Portion) in Accordance with 10 CFR 50.55a(f)(4)(iv)
Proposed Subsequent Code Edition and Addenda (Portion)

JAF proposes to use a portion of a subsequent edition of the ASME OM Code. Specifically, ASME OM Code 2004 Edition no addenda, Appendix I, Paragraph I-3410(d). This requirement will be applied to 02RV-71A, B, C, D, E, F, G, H, J, K, L.

Related Requirements

Although changes were issued to other paragraphs in the 2004 Edition of the OM Code, review of these changes identified no related requirements for ASME OM Code 2004 Edition, Appendix I, Paragraph I-3410(d).

Duration:

The proposed alternative and use of a subsequent code edition identified in this 10 CFR 50.55a Request shall be utilized during the Fourth Ten year IST Interval that began on October 1, 2007.

Precedents:

NUREG-1482 Paragraph 4.3.2.1 states, "In recent years, the NRC staff has received numerous requests for relief and/or TS changes related to the stroke testing requirements for BWR dual-function main steam SRVs. Both Appendix I to the ASME OM Code and the plant-specific TS require stroke testing of SRVs after they are reinstalled following maintenance activities. Several licensees have determined that in situ testing of the SRVs can contribute to undesirable seat leakage of the valves during subsequent plant operation and have received approval to perform testing at a laboratory facility coupled with in situ tests and other verifications of actuation systems as an alternative to the testing required by the ASME OM Code and TS."

References:

1. ASME OM Code, 2001 Edition through 2003 Addenda
2. ASME OM Code, 2004 Edition no Addenda
3. NUREG-1482 Rev. 1, "Guidelines for Inservice Testing at Nuclear power Plants"
4. JAF UFSAR 4.4.7 Pressure Relief System Inspection and Testing