UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 29, 2016

Vice-President, Operations
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – INSERVICE INSPECTION PROGRAM ALTERNATIVE FOR SAFETY RELIEF VALVES (CAC NO. MF8013)

Dear Sir or Madam:

By letter dated June 21, 2016, as supplemented by letter dated June 27, 2016, Entergy Nuclear Operations, Inc. (the licensee) submitted an inservice testing alternative to the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section XI, Subsection IWB-5221, for James A. FitzPatrick Nuclear Power Plant (JAF). The alternative requested U.S. Nuclear Regulatory Commission (NRC) approval to perform the VT-2 visual examination during a system leakage test of Class 1 components with mechanical joint connections at a pressure lower than the ASME Code-required pressure following the replacement of main steam safety relief valves.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that complying with the ASME Code requirement would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Furthermore, the NRC staff has determined that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the subject components. Therefore, pursuant to Title 10 of the *Code of Federal Regulations*, Section 50.55a(z)(2), the NRC staff authorizes the use of the alternative for JAF until its scheduled permanent cessation of operation on January 27, 2017.

All other ASME Code, Section XI, requirements for which relief or an alternative was not specifically requested and approved in the subject request remains applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, regarding this matter, please contact the NRC Project Manager, William Huffman, at 301-415-2046 or via e-mail at William.Huffman@nrc.gov.

Sincerely,

Shaun M. Anderson, Acting Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST RR-20 REGARDING SYSTEM LEAKAGE TEST

OF MECHANICAL JOINT CONNECTIONS

ENTERGY NUCLEAR OPERATIONS, INC.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated June 21, 2016 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML16173A465), as supplemented by letter dated June 27, 2016 (ADAMS Accession No. ML16179A439), Entergy Nuclear Operations, Inc. (the licensee) requested relief from the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section XI, Subsection IWB-5221, for James A. FitzPatrick Nuclear Power Plant (JAF). The licensee proposed to perform a system leakage test of Class 1 pressure-retaining mechanical joint connections at a lower pressure than the ASME Code-required pressure following repair and replacement activities.

Specifically, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(z)(2), the licensee proposed an alternative in Relief Request RR-20 to the pressure test requirements of ASME Code, Section XI, Subsection IWB-5221(a), for repair/replacement activities of Class 1 mechanical joint connections made in the installation of pressure-retaining items.

2.0 REGULATORY EVALUATION

The regulation at 10 CFR 50.55a(g)(4) states that ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of components.

The regulation at 10 CFR 50.55a(z) states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if the licensee demonstrates that (1) the proposed alternatives would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Enclosure

Based on the above, and subject to the following technical evaluation, the U.S. Nuclear Regulatory Commission (NRC) staff finds that regulatory authority exists for the licensee to request the use of the alternative and the NRC to authorize the proposed alternative.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components Affected

Class 1 pressure-retaining mechanical joint connections that require a system leakage test with associated VT-2 visual examination subsequent to repair/replacement activities.

3.2 Applicable ASME Code

ASME Section XI Code, 2001 Edition through 2003 Addenda.

3.3 ASME Code and 10 CFR 50.55a Requirements

IWA-4540(a) requires, unless exempted by IWA-4540(b), a system leakage test in accordance with IWA-5000 for repair/replacement activities performed by welding or brazing on a pressure-retaining boundary prior to, or as part of, returning to service.

IWA-5213(b) requires a 10-minute hold time for noninsulated components and a 4-hour hold time for insulated components prior to performing the VT-2 leakage test.

IWB-5221(a) requires the system leakage test to be conducted at a pressure not less than the nominal pressure associated with 100 percent rated reactor power.

Section 50.55a(b)(2)(xxvi) of 10 CFR, "Section XI condition: Pressure Testing Class 1, 2, and 3 Mechanical Joints," provides supplemental requirements to those of IWA-4540(a) stated above. Section 50.55a(b)(2)(xxvi) of 10 CFR requires the use of repair/replacement activity provisions of IWA-4540(c) in the 1998 Edition of Section XI for pressure testing of Class 1, 2, and 3 mechanical joints when using the 2001 Edition through the latest edition and addenda of ASME Code, Section XI. Therefore, even though the inservice inspection Code of Record applicable at JAF does not require pressure testing and VT-2 examination of mechanical joint connections, the 1998 Edition of Section XI does.

3.4 Reason for Request

The licensee is requesting an alternative to the requirements of ASME Code Section XI, 2001 Edition through the 2003 Addenda, Subsection IWB 5221(a), which requires that the system leakage test be conducted at a test pressure not less than the normal operating pressure associated with 100 percent power, which for JAF is 1040 pounds per square inch gauge (psig)

The licensee plans to replace some components installed via mechanical joints (e.g., main steam safety relief valves (SRVs)) during an ongoing unplanned shutdown that started on June 24, 2016. The SRV replacement activities will require a system leakage test and associated VT-2 visual examination of the mechanical joint connections.

The licensee requests NRC approval of the proposed alternative to perform the VT-2 visual examination during a system leakage test of Class 1 components with mechanical joint connections at a lower pressure than the ASME Code-required pressure following repair and replacement activities. This proposed alternative would allow the performance of the VT-2 visual leakage examination following the SRV repair and replacement activities at the lower pressure of greater than or equal (≥) to 905 psig. The licensee stated that it will perform the system leakage test at a test pressure of at least 55 pounds per square inch above 905 psig (905 psig is the calculated minimum test pressure required by ASME Code Case of at least 87 percent of the pressure required by IWB-5221(a)).

Performance of a cold leakage test (that is, a non-nuclear heatup such as that required following a refueling outage) subsequent to the current shutdown is judged to be an imprudent course of action for the reasons identified in Section 3.5.5. of this safety evaluation. Performance of VT-2 inspections in the drywell at 1040 psig during startup, while raising pressure set with the main turbine offline, could result in stability/controllability concerns. The licensee states that this alternative is therefore justified since compliance with the cited requirements of the identified code would result in a plant hardship without a compensating increase in the level of quality and safety of the associated maintenance activity.

3.5 Proposed Alternative

The licensee proposed to perform the required system leakage test and associated VT-2 visual examination for any repair/replacement activities of mechanical joint connections upon startup from the current outage using nuclear heat (steam) at a reactor pressure of at least 960 psig. The licensee proposed to implement a 1-hour hold time for noninsulated components and a 6-hour hold time for insulated components prior to performing the VT-2 visual examination. If there are additional unplanned shutdowns with drywell entries before permanent cessation of operation, the licensee proposed to perform an initial walkdown upon entry to identify any evidence of leakage from SRV-related repair and replacement activities.

The licensee stated that because the RCS pressure boundary is subjected to a leakage test and visual examination at nominal operating pressure (i.e., 1,040 psig) near the end of every refueling outage, and monitoring systems detect leakage inside the drywell, a leakage test and visual examination performed at a pressure of at least 960 psig for the repair/replacement of mechanical joint connections provide adequate assurance of structural and pressure boundary integrity.

3.5.1 Test Pressure

The licensee stated that it will perform the required VT-2 visual examination for any repair/replacement activities of mechanical joint connections performed during the current outage or any future shutdown at a reactor pressure of ≥ 905 psig. The test pressure of at least 905 psig is consistent with ASME Code Case N-795, "Alternative Requirements for BWR [Boiling-Water Reactor] Class 1 System Leakage Test Pressure Following Repair/Replacement Activities, Section XI, Division 1," which requires that the test pressure be at least 87 percent of the pressure corresponding to 100 percent of rated power (i.e., 1040 psig). In addition, the licensee stated that it would perform the system leakage test at a test pressure of at least 5 percent (at least 55 psi) above the minimum test pressure of 905 psig (i.e., at 960 psig) to provide additional margin.

3.5.2 System Leakage Test

The licensee proposed to perform the system leakage test using nuclear heat (steam) as part of the plant startup sequence. In its supplement to the application dated June 27, 2016, the licensee provided a comparison of the proposed system leakage test to the normal system leakage test to show that the proposed system leakage test provides safer plant operation than the normal system leakage test. During the normal system leakage test, some emergency core cooling systems are either isolated or on standby, and therefore, would not be able to respond quickly to protect the reactor, if needed. The reactor pressure vessel (RPV) and main steam lines would be water solid during a normal system leakage test, making it difficult to control pressure in that configuration. In addition, a system leakage test with low temperature and high pressure places the plant closer to the RPV pressure-temperature (P-T) limits, which is undesirable in terms of safe plant operation. As shown in the table below, during the proposed system leakage test, important emergency core cooling systems are operable to protect the reactor. The proposed system leakage test will not affect the P-T limits of the RPV, so as to protect the fracture toughness of the plates and welds of the RPV shell.

	System Leakage Test during startup with RR-20	"Hydro" Test (ST-39H)
Medium	Steam (nuclear heat)	Water
Pressure Control	Electro-Hydraulic Control (EHC) System	Difficult to control pressure in water solid state
RPV P-T Limits	Not affected	Approaching P-T limits
High Pressure Coolant injection	Operable	Isolated
Core Spray	Operable	Operable
Low Pressure Coolant Injection	Operable	Operable
Extensive valve manipulations	No (only per plant startup)	Yes per ST-39H
Reactor Vessel and Main Steam Lines are virtually water solid	No	Yes
Reactor Core Isolation Cooling System	Operable	Isolated
Extensive water moves	No	Yes
Flood Risk	Steam will be present at a lower than operating pressure	Water at a higher pressure present therefore a greater flood risk
Energy input to the drywell if leakage is detected after replacement	More energy input due to higher steam temperature	Less energy input due to lower temperature water

3.5.3 Hold Time

As part of the proposed alternative, the licensee will implement a 1-hour hold time for noninsulated components and a 6-hour hold time for insulated components prior to performing the VT-2 visual examination.

3.5.4 Additional Monitoring

The licensee stated that the disposition of any observed leakage will consider the marginal increase in leakage rates that might occur at the nominal operating pressure associated with 100 percent rated reactor power (i.e. 1040 psig) and the actual reactor pressure when the examination was performed. The licensee recognized that the Class 1 pressure boundary leakage rates detected at a lower test pressure would likely be greater if detected at the higher nominal operating pressure associated with 100 percent rated power (1040 psig). The licensee stated that it would consider the higher leakage rates at higher pressures during evaluations for corrective action should leakage be detected, and that it will follow the ASME Code, Section XI, IWA-5250(a). Any deficiencies will be evaluated in the licensee's Corrective Action Program. In the case of identified leakage, a ratio comparing the lower test pressure to the higher operating pressure would be used during engineering evaluations of identified leakage.

The licensee noted that drywell monitoring systems would detect leakage that might occur in mechanical joint connections at higher pressures associated with nominal reactor operation. These systems include drywell air temperature and pressure monitoring and the drywell floor and equipment drain sumps.

According to the licensee, because the reactor coolant system pressure boundary is subjected to a leakage test and visual examination at nominal operating pressure (i.e., 1040 psig) near the end of every refueling outage and monitoring systems detect leakage inside the drywell, a leakage test and visual examination performed at or above 905 psig for the repair/replacement of mechanical joint connections provides adequate assurance of structural and pressure boundary integrity.

The licensee stated that if there are additional unplanned shutdowns with drywell entries before permanent cessation of operation, it will perform an initial walkdown upon entry to identify any evidence of leakage from SRV-related repair and replacement activities.

3.5.5 Hardship and Unusual Difficulty Justification

The licensee requested relief from the test pressure requirement of IWB-5221(a) (i.e., 1040 psig) on the basis of hardship, without a compensating increase in the level of quality and safety, as cited below.

The licensee stated that the VT-2 visual examination inside the drywell (primary containment) represents a hardship at the nominal 100 percent power operating pressure of 1040 psig during startup because of drywell entry restrictions above 15 percent power and prohibitively high dose rates. The JAF Final Safety Analysis Report, Section 5.2.3.4, states, in part, that "[d]rywell entry is limited to conditions where the reactor power is fifteen percent or less of rated thermal power...."

The licensee further stated that raising the test pressure to 1040 psig for VT-2 visual examinations in the drywell, with the main turbine offline, would represent a high risk because the plant would be in a configuration where controllability could be an issue due to responsiveness to small power or pressure perturbations and operating near the scram set point.

The licensee noted that performance of a cold leakage test (that is, a non-nuclear heatup such as that required following a refueling outage) subsequent to a maintenance shutdown is judged to be an imprudent course of action for the reasons described below:

- Main Steam Lines are flooded with the main steam isolation valves closed.
- The RPV is required to be virtually water solid.
- Performance of an additional cold leakage test places the unit in a position of reduced margin, unnecessarily approaching the fracture toughness limits defined in the Technical Specification P–T curves.
- Extensive valve manipulations, system lineups, and procedural controls are required in order to heat up and pressurize the reactor coolant system to establish the necessary test pressure.
- The additional valve lineups and system reconfigurations necessary to support this test will impose an additional challenge to the affected systems.
- Performing a cold leakage test would add approximately 2 days to the shutdown duration.
- The scope of the VT-2 leakage examination does not include the RPV.

3.6 Duration of Proposed Alternative

The licensee stated that NRC approval is sought prior to startup from the current outage, with the alternative continuing in effect through the end of plant life January 27, 2017.

3.7 NRC Staff Evaluation

The NRC staff evaluated the licensee's technical basis for the proposed test pressure, system leakage testing, hold time, additional monitoring, and hardship and usual difficulty justification to determine the acceptability of the proposed alternative.

3.7.1 Test Pressure

During the current outage (which commenced on June 24, 2016), the licensee is replacing the main steam SRVs. The ASME Code, Section XI, IWA-4540(a), of the 2001 Edition through 2003 Addenda, requires a system leakage test for repair/replacement activities of Class 1, 2, and 3 components performed by welding or brazing on a pressure-retaining boundary prior to, or as part of, returning to service. However, IWA-4540(b) exempts mechanical joints from system leakage testing. The NRC staff notes that replacement of an SRV involves mechanical joint connections. Therefore, based on IWA-4540(b) of the 2001 Edition through 2003 Addenda, the licensee is not required to perform a system leakage test after replacing SRVs at JAF.

However, 10 CFR 50.55a(b)(2)(xxvi) prohibits the use of ASME Code, Section XI, 2001 Edition, and later editions and addenda, when performing system leakage tests. This limitation requires that the repair and replacement activity provisions of ASME Code, Section XI, IWA-4540(c), of the 1998 Edition of ASME Code, Section XI, for pressure testing Class 1, 2, and 3 mechanical joints be applied. Article IWA-4540(c) of the 1998 Edition requires that mechanical joints made in the installation of pressure-retaining items be pressure tested in accordance with ASME Code, Section XI, IWA-5211(a).

The 1998 Edition of ASME Code, Section XI, IWA-5211(a), describes the system leakage test while the system is in operation, during a system operability test, or while the system is at test conditions using an external pressurization source.

The 1998 Edition of ASME Code, Section XI, IWB-5220, requires a system leakage test specifically for ASME Class 1 components, such as SRVs. ASME Code, Section XI, IWB-5221(a), requires a system leakage test to be performed at a pressure not less than the pressure corresponding to 100 percent rated reactor power (i.e., 1,040 psig). The licensee requested relief from IWB-5221(a) of the 2001 Edition through 2003 Addenda of ASME Code, Section XI, and proposed to deviate from the test pressure of 1,040 psig as required by IWB-5221(a) of the 1998 Edition of the ASME Code, Section XI.

The *Federal Register* Notice (69 FR 58804), dated October 1, 2004, which incorporated Section 50.55a(b)(2)(xxvi) of 10 CFR, provided the basis for the system leakage test requirement. The NRC staff believed the system pressure testing was necessary to ensure and verify the integrity of the pressure boundary. The *Federal Register* Notice did not discuss specific test pressure values but focused on the need to perform a pressure test and VT-2 visual examination to verify the integrity of the pressure boundary after repair and replacement activities.

As an alternative to the test pressure required in IWB-5221(a), the licensee proposed to perform the system leakage test at a pressure of 960 psig, which is 5 percent higher than the required minimum pressure of 905 psig per the requirement of ASME Code Case N-795. The NRC staff determines that using a test pressure of 960 psig, the licensee satisfies the minimum test pressure requirement of the ASME Code Case N-795.

Based on the above evaluation, the NRC staff finds that the proposed test pressure of at least 960 psig is sufficiently high to cause detectable leakage from any mechanical joint connections following disassembly and reassembly of the affected SRVs, if leakage occurs.

Based on the above evaluation, the NRC staff finds that a test pressure of at least 960 psig for the system leakage test would not significantly reduce plant safety and, therefore, is acceptable.

3.7.2 System Leakage Test

The NRC staff finds that the licensee has demonstrated that for pressure testing following repair and replacement activities conducted during short maintenance outages, using the nuclear heat to perform the proposed system pressure test provides a safer and less risky plant operation than the normal system leakage testing. The NRC staff further finds that during the proposed system leakage test, important emergency core cooling systems are operable whereas in the normal system leakage test, they are either isolated or on standby.

The NRC staff notes that the proposed system leakage test will not approach the RPV P-T fracture toughness limits associated with the welds and plates of the RPV shell, whereas the normal system leakage test could challenge the P-T limits. The NRC staff further notes that during a typical post refuel outage system leakage test, the reactor vessel and main steam lines are essentially water solid and safety systems are isolated or in standby. The alternate method of using nuclear steam (heat) to obtain the required test pressure will use the Electro-Hydraulic Control System to control pressure. This is a normal pressure control mode for a BWR RPV temperature and level in more normal operating bands. Safety systems will also be operable during this leakage test.

Therefore, the NRC staff finds that the proposed system leakage test provides a safer plant operation and lower risk as compared to the normal system leakage test.

3.7.3 Hold Time

The licensee proposed to implement a 1-hour hold time for non-insulated components and a 6-hour hold time for insulated components prior to performing the VT-2 visual examination. The requirements in IWA-5213(a) specify a 10-minute hold time for non-insulated components and a 4-hour hold time for insulated components prior to performing the VT-2 visual examination. The NRC staff finds that the increased hold times are more conservative than the requirements in IWA-5213(a) because longer hold times increase the possibility of observing leakage, if it occurs. Therefore, the NRC concludes that the proposed hold time is acceptable.

3.7.4 Additional Monitoring

The NRC staff notes that if there are unplanned shutdowns with drywell entries before permanent cessation of operation on January 27, 2017, the licensee will perform an initial walkdown upon entry to identify any evidence of leakage from SRV-related repair and replacement activities. The NRC staff finds that this additional monitoring will increase the ability to discover any SRV leakage if it exists and, therefore, is acceptable.

The NRC staff notes that JAF has drywell air temperature and pressure monitors and the drywell floor and equipment drain sumps to detect leakage that might occur in mechanical joint connections. The NRC staff finds that these detection systems will detect potential leakage and thereby enhance the monitoring of the structural integrity and leak-tightness of the replacement SRVs.

The licensee recognizes that Class 1 pressure boundary leakage rates detected at the requested lower test pressure would likely be greater if detected at the higher nominal operating pressure associated with 100 percent rated power (1040 psig). The recognition of higher leakage rates at higher pressures would be taken into consideration during evaluations for corrective action should leakage be detected. The licensee stated that it will follow IWA-5250(a) and any deficiencies will be evaluated in the Corrective Action Program. In the case of identified leakage, a ratio comparing the lower test pressure to the higher operating pressure would be used during engineering evaluations of identified leakage.

The NRC staff finds this approach acceptable because the proposed minimum test pressure is close to the normal operating pressure. Due to this minimal pressure difference, an extrapolation of the test leakage rate, if any leakage occurs, should be close to the leakage rates from a system leakage test that uses the pressure at 100 percent rated power.

3.7.5 Hardship and Unusual Difficulty Justification

To perform the system leakage test with the associated VT-2 visual examination inside the drywell, in accordance with ASME Code, Section XI, IWB-5221(a) requirements, is a hardship at the nominal 100 percent operating pressure of 1040 psig during startup. This is because the drywell entry is restricted when the reactor power approaches the rated thermal power and dose rates are prohibitive. The NRC notes that raising the test pressure to 1040 psig for VT-2 visual examination in the drywell, with the main turbine offline, could be a high risk as the plant would be in a configuration where controllability could be an issue due to responsiveness to small power or pressure perturbation and operating near the scram set point. Based on the above, the NRC staff finds that performing the VT-2 visual examination in accordance with the ASME Code is an unusual difficulty for plant personnel.

As an alternative to satisfy ASME Code, Section XI, IWB-5221(a), the licensee could perform a cold leakage test based on a non-nuclear heatup. Performing this type of test would require filling the main steam lines and the reactor vessel solid with water, which is undesirable for system operation. To establish the necessary test conditions, the licensee would have to perform valve lineups, and system manipulations, all of which activities would require additional personnel radiation exposure beyond that received in normal startup activities. Additionally, these activities may introduce human errors. The cold leakage test places the unit in a position of reduced margin, unnecessarily approaching the fracture toughness limits defined in the technical specification P-T curves. Based on the above, the NRC staff finds that performing a cold leakage test would cause a hardship for the licensee.

4.0 CONCLUSION

As set forth above, the NRC staff has determined that complying with the ASME Code requirement would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Furthermore, the NRC staff has determined that the proposed Relief Request RR-20 provides reasonable assurance of structural integrity and leak tightness of the subject components. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in

10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the proposed alternative in accordance with 10 CFR 50.55a(z)(2) for JAF through the end of plant life on January 27, 2017.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: John Tsao

Date: June 29,2016

If you have any questions, regarding this matter, please contact the NRC Project Manager, William Huffman, at 301-415-2046 or via e-mail at William.Huffman@nrc.gov.

Sincerely,

/RA/

Shaun M. Anderson, Acting Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosure:
Safety Evaluation

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