Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
James A. FitzPatrick NPP
P.O. Box 110
Lycoming, NY 13093
Tel 315-342-3840

William C. Drews
Regulatory Assurance Manager – JAF

JAFP-16-0108
June 21, 2016

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: Proposed Inservice Inspection Program Alternative in Accordance with 10 CFR 50.55a(z)(2), RR-20 at James A. FitzPatrick Nuclear Power Plant

James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-59

Reference: NRC to Southern Nuclear Operating Company, Inc., Edwin I. Hatch Nuclear Plant, Unit No. 2 – Inservice Inspection Program Alternative for Safety Relief Valves (CAC No. MF7692), ML16134A119, dated May 16, 2016

Dear Sir or Madam:

Pursuant to 10 Code of Federal Regulations (CFR) 50.55a(z)(2), James A. FitzPatrick Nuclear Power Plant (JAF) hereby requests approval of an alternative to specific portions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” on the basis that the proposed alternative provides an acceptable level of quality and safety.

JAF 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% power, which for JAF is 1040 psig. These pressure tests requirements are supplemented by 10 CFR 50.55a(b)(2)(xxvi) which invokes the requirements of IWA 4540(c) of the 1998 Edition of ASME Section XI Code for repair/replacement activities of Class 1, 2, and 3 mechanical joint connections. JAF is currently in the fourth 10-year Inservice Inspection (ISI) interval, which began on March 1, 2007 and ends February 3, 2017. The ISI Code of Record for the fourth interval is ASME Section XI, 2001 Edition through the 2003 Addenda.

Specifically, JAF requests NRC approval of proposed alternative Relief Request RR-20, 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 Code-required pressure following repair and replacement activities. This proposed alternative would allow the performance of the

VT-2 visual leakage examination following Main Steam Safety Relief Valve repair and replacement activities at the lower pressure of greater than or equal to 905 psig while employing a one hour hold time for non-insulated components and a six hour hold time for insulated components. Additionally, JAF would intend to perform the system leakage test at a test pressure of at least 55 psig above 905 psig [(905 psig is the calculated minimum test pressure required by ASME Code Case N-795 (at least 87% of the pressure required by IWB-5221(a))].

10 CFR 50.55a(z)(2) provides for the use of such alternatives to the requirements of paragraph 50.55a(g) when so authorized by the NRC. Such authorization is contingent upon a demonstration by the applicant that the proposed alternative would provide an acceptable level of quality and safety or that the specified requirements would result in hardship or unusual difficulty without a commensurate increase in the level of quality and safety. Performance of a cold leakage test (that is, a non-nuclear heat-up 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 identified in the Attachment. 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. This alternative is therefore justified since compliance with the cited requirements of the identified code would result in a plant hardship without a commensurate increase in the level of quality and safety of the associated maintenance activity. Further details of the demonstration required by 10 CFR 50.55a(z)(2) are provided in the Attachment to this letter.

The NRC has previously approved several similar relief requests for performing pressure tests at less than nominal operating pressure including Southern Nuclear Operating Company's recent request for approval of ISI Program Alternative HPN-ISI-ALT-05-02 (Reference).

JAF requests NRC Staff review and approval of this proposed alternative on or before June 28, 2016 to accommodate application of this request during the next potential outage.

There are no new regulatory commitments made in this letter. Should you have any questions, please contact the Regulatory Assurance Manager, Mr. William C. Drews, at (315) 349-6562.

Very truly yours,

M. Hawes acting for

William C. Drews
Regulatory Assurance Manager

WCD:dc

Attachment: James A. FitzPatrick Nuclear Power Plant Inservice Inspection Program Alternative Relief Request RR-20

cc: USNRC, Regional Administrator, Region I
USNRC, Project Directorate
USNRC, Resident Inspector

JAFP-16-0108

Attachment

James A. FitzPatrick Nuclear Power Plant
Inservice Inspection Program Alternative Relief Request RR-20
(4 Pages)

James A. FitzPatrick Nuclear Power Plant
Inservice Inspection Program Alternative Relief Request

RR-20

Plant Site-Unit:	James A. FitzPatrick Nuclear Plant
Interval-Interval Dates:	4th ISI Interval, March 1, 2007 through February 3, 2017
Requested Date for Approval:	Approval is requested by June 28, 2016 to support testing during a maintenance shutdown that may begin after June 22, 2016.
ASME Code Components Affected:	Class 1 pressure-retaining mechanical joint connections which require a VT-2 examination for leakage subsequent to repair/replacement activities.
Applicable Code Edition and Addenda:	ASME Section XI Code, 2001 Edition through the 2003 Addenda
Applicable Code Requirements:	1. IWA-4540(a) unless exempted by IWA-4540(b) a system leakage test, in accordance with IWA-5000, is required for repair/replacement activities performed by welding or brazing on a pressure-retaining boundary prior to, or as part of, returning to service. 2. IWA-5213(b) requires a 10 minute hold time for non-insulated components and 4 hour hold time for insulated components prior to performing the VT-2 leakage test. 3. IWB-5221(a) requires the system leakage test to be conducted at a pressure not less than the nominal pressure associated with 100% rated reactor power.
Reason for Request:	10 CFR 50.55a(b)(2)(xxvi) *Pressure Testing Class 1, 2, and 3 Mechanical Joints* provides supplemental code requirements to those of IWA-4540(a) stated above. 10 CFR 50.55a(b)(2)(xxvi) invokes the IWA-4540(c) repair/replacement activity provisions of 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 Section XI. Therefore, even though the ISI 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. Relief is requested from the test pressure requirement of IWB-5221(a) (i.e., 1040 psig) on the basis of hardship as cited below. • Replacement of some components installed via mechanical joints (e.g., Safety Relief Valves (SRVs)) is planned during a shutdown which may begin after June 22, 2016. These repair/replacement activities will require a VT-2 leakage examination of the mechanical joint connections during unit startup.

	• VT-2 leakage examination inside the drywell (primary containment) represents a hardship at the nominal operating pressure of 1040 psig when reached at 100% power during start-up because of Drywell entry restrictions above 15% power and dose rate being prohibitive to entry. o FSAR Section 5.2.3.4 Drywell entry is limited to conditions where the reactor power is fifteen percent or less of rated thermal power • Nominal operation pressure (i.e., 1040 psig) will not be reached for more than 24 hours after reaching 905 psig during the startup sequence. • Raising the pressure set to 1040 psig for VT-2 inspections in the drywell, with the main turbine offline, would be high risk as the plant would be in a place where controllability could be an issue due to responsiveness to small power or pressure perturbation and being near the scram set point. • Performance of a cold leakage test (that is, a non-nuclear heat-up 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: o Main Steam Lines are flooded with Main Steam Isolation Valves closed. o The reactor pressure vessel (RPV) is required to be virtually water solid. o 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 Pressure-Temperature (P–T) curves. o 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. o The additional valve lineups and system reconfigurations necessary to support this test will impose an additional challenge to the affected systems. A normal plant startup would then occur, after completion and subsequent recovery from the cold leakage test. o Performing a cold leakage test would add approximately 2 days to the shutdown duration. o The scope of the VT-2 leakage examination does not include the reactor pressure vessel.
Proposed Alternative and Basis for Use:	JAF may perform the required VT-2 leakage examination for any repair/ replacement activities of mechanical joint connections performed in a future shutdown at a reactor pressure of ≥ 905 psig (consistent with 87% of the pressure required by IWB-5221(a) from ASME Code Case N-795). Similar to submittals by Hatch, JAF agrees 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 Leakage Test. Disposition of any observed leakage will consider the marginal increase in

James A. FitzPatrick Nuclear Power Plant
Inservice Inspection Program Alternative Relief Request

RR-20

	leakage rates that might occur at the nominal operating pressure associated with 100% rated reactor power (i.e. 1040 psig) and the actual reactor pressure when the examination was performed. In addition, 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. Since 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 set up and performance of a system pressure test for a limited number of system joints would result in hardships associated with a plant nearing the end of its scheduled operational life that has historically demonstrated a low reactor coolant system unidentified leakage rate without a compensating increase in quality or safety. The proposed alternative provides an acceptable level of quality and safety and is consistent with requirements of ASME Code Case N-795 for a BWR Class 1 system leakage test, following repair/replacement activities. JAF proposes to perform the system leakage test at a pressure at least 5% (at least 55 psig) above the minimum pressure required by Code Case N-795. Therefore, this proposed alternative should be granted pursuant to 10 CFR 50.55a(z)(2). This alternative is similar to Alternatives HNP-ISI-ALT-09, Version 2.0, and HNP-ISI-ALT-15, Version 2.0, both previously approved by the NRC for Hatch Unit 2 (References 3 and 4, respectively).
Duration of Proposed Alternative:	Upon approval, as JAF may shutdown prior to January 27, 2017, with the alternative continuing in effect through the end of plant life January 27, 2017.
References:	1. Nuclear Management Company, Monticello Nuclear Generating Plant, 3rd 10-Year Interval ISI Program Relief Request RR-17, NRC TAC NO. MC0593 dated March 25, 2004. 2. PSEG Nuclear, LLC, Hope Creek Nuclear Generating Station, 2nd 10-Year Interval ISI Program Relief Request HC-RR-12-023, NRC TAC No. MC2396 dated August 27, 2004. 3. Southern Nuclear Operating Company, Inc., Edwin I. Hatch Plant Unit 2, ISI Program Alternative HNP-ISI-ALT-09 Version 2.0, March 29, 2010 (NRC ADAMS Accession No. ML100890051). 4. Southern Nuclear Operating Company, Inc., Edwin I. Hatch Plant Unit 2, ISI Program Alternative HNP-ISI-ALT-15 Version 2.0, December 13, 2011 (NRC ADAMS Accession No. ML113480294).

James A. FitzPatrick Nuclear Power Plant
Inservice Inspection Program Alternative Relief Request

RR-20

	5. NextEra Energy Duane Arnold, LLC, Duane Arnold Energy Center, Request for Authorization of Alternative Regarding Pressure Test Requirements, NRC TAC No. ME5143 dated September 6, 2011.
Status:	Awaiting NRC approval.