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Subject: Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1), Implementation of BWRVIP-05 (GL 98-05) at James A. FitzPatrick Nuclear Power Plant

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

Dear Sir or Madam:

Pursuant to 10 Code of Federal Regulations (CFR) 50.55a(z)(1), James A. FitzPatrick Nuclear Power Plant (JAF) hereby requests an alternative to specific portions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for In-service 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 ASME Section XI, 2001 Edition through the 2003 Addenda, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, which requires volumetric examination of the reactor vessel circumferential shell welds each in-service inspection (ISI) interval. JAF is currently in the fourth 10-year ISI interval, which began on March 1, 2007 and ends December 31, 2016. The ISI Code of Record for the fourth interval is ASME Section XI, 2001 Edition through the 2003 Addenda.

JAF requests NRC Staff review and approval of this proposed alternative on or before September 1, 2016 to accommodate application of this request during the next refueling outage.

There are no commitments made in this letter. Should you have any questions, please contact the Regulatory Assurance Manager, Mr. Chris M. Adner, at (315) 349-6766.

Very truly yours,

Chris M. Adner
Regulatory Assurance Manager

CMA:ds

Enclosure 1: James A. FitzPatrick Nuclear Power Plant In-service Inspection Program RR-19

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

JAFP-15-0108

Enclosure 1

James A. FitzPatrick Nuclear Power Plant
In-service Inspection Program RR-19
(8 pages)

Entergy Nuclear Operations, Inc
James A. FitzPatrick Nuclear Power Plant
Proposed Alternative in Accordance with 10CFR50.55a(z)(1)
Implementation of BWRVIP-05 (GL 98-05) – Relief Request No. RR-19

1. ASME Code Component(s) Affected

Code Class:	ASME Section XI Code Class 1
Component Numbers:	VC-1-2, VC-2-3, VC-3-4, VC-4-BH-1
Code References:	ASME Section XI, 2001 Edition with 2003 Addenda
	BWRVIP-05: BWR Vessel and Internals Project: "BWR Reactor Pressure Vessel Shell Inspection Recommendations."
Examination Category:	B-A
Item Number(s):	B1.11
Unit/Inspection Interval:	James A. FitzPatrick (JAF) / Fourth (4^{th}) 10-year interval starting March 1, 2007, ending December 31, 2016 and including the Period of Extended Operation.

2. Applicable ASME Code Requirements

ASME Section XI, 2001 Edition through the 2003 Addenda, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, requires a volumetric examination of the circumferential shell welds each interval.

3. Reason for Request

During the review of the James A. FitzPatrick Nuclear Power Plant (JAF) License Renewal Application (LRA), in Request for Additional Information (RAI) 4.2.5-1 dated January 12, 2007 the staff asked JAF whether it intended to apply for relief from the ASME Code Section XI circumferential weld examination requirements for the period of extended operation.

The JAF response dated February 12, 2007 indicated that a request for alternative under the provisions of 10 CFR 50.55a would be submitted to exclude the Reactor Pressure Vessel (RPV) shell circumferential welds from examination. Based on Entergy's response, the RAI was considered resolved.

As such, JAF is requesting an alternative in accordance with 10 CFR 50.55a(z)(1) on the basis that this alternative provides an acceptable level of quality and safety. This request for alternative would provide relief from circumferential weld examinations required by the ASME Section XI Code for the period of extended operation.

JAF was previously granted this relief for the remainder of the original 40-year license term (Reference TAC No. MA6215, Accession No. ML003685801).

4. Proposed Alternative

JAF requests to use BWRVIP-05 (Reference 2), with supporting information described herein, as the bases for excluding the RPV shell circumferential welds from the examination required by ASME Section XI, Examination Category B-A, Item No. B1.11 for the extended license period ending on October 17, 2034.

The axial weld seams (Examination Category B-A, Item No. B1.12) and their intersection with the associated circumferential weld seams will be examined in accordance with ASME Section XI except where specific relief is granted when essentially 100% (>90%) coverage cannot be obtained.

5. Basis for Use

The technical basis supporting the requested alternative is provided by BWRVIP-05, (EPRI TR-105697) "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations" (Reference 2) as accepted in the staff's final safety evaluation report enclosed in a July 28, 1998, letter from Mr. G.C. Lanais, NRC, to Mr. C. Terry, the BWRVIP Chairman (Reference 3). In this letter, the staff concluded that because the failure frequency for circumferential welds in BWR plants is significantly below the criterion specified in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," (Reference 9) and below the core damage frequency of any BWR plant, continued inspection would result in a negligible decrease in an already acceptably low RPV failure probability and justify elimination of the In-Service Inspection (ISI) requirements for RPV circumferential welds. The staff's letter indicated that BWR applicants may request relief from ASME Code Section XI requirements for volumetric examination of circumferential RPV welds by demonstrating that (1) at the expiration of the license the circumferential welds will satisfy the staff's July 28, 1998, evaluation of the limiting conditional failure probability for circumferential welds and (2) the applicants have implemented operator training and established procedures that limit the cold over-pressure event frequency to that specified in the staff's Safety Evaluation Report (SER). The letter also indicated that the requirements for inspection of RPV circumferential welds during an additional 20-year license renewal period would need plant-specific reassessment as part of any BWR LRA. The applicant also must request relief from the ASME Code Section XI requirements for volumetric examination of circumferential welds for the extended license term in accordance with 10 CFR 50.55a(z)(1).

Additional information to support the proposed alternative for the license renewal period is contained in the JAF LRA and its amendments as accepted by the staff in their SER dated April 2008 (Reference 8). The one exception requiring further action by JAF affecting the Reactor Vessel Neutron Fluence Time-Limited Aging Analysis (TLAA) was license condition 4.2.1-1 which has been closed.

In accordance with the guidance of Generic Letter 98-05, applicants requesting relief from examination of the subject welds are to demonstrate that: (1) at the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 30, 1998 safety evaluation, and (2) licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 30, 1998 safety evaluation.

(1) Satisfying the Limiting Conditional Failure Probability for Circumferential Welds

Section 4.2.5.2 of the SER (Reference 8) for the LRA states in part:

> The results in SER Table 4.2.5-1 on the following page demonstrate that the mean RT_{NDT} value calculated by the applicant for the reactor vessel circumferential weld is less than that for the limiting CEOG case study and agrees with that calculated by the staff. Based on this analysis, the staff concludes that the applicant has provided a valid basis for the conclusion that the conditional probability of failure for the reactor vessel circumferential weld is sufficiently low to accept the TLAA and set the basis for a relief request to eliminate the reactor vessel circumferential weld examinations for the period of extended of operation after renewal of the operating license.

Table 1 and corresponding notes excerpted from SER for the LRA, section 4.2.5.2:

> Table 1 shows a comparison between the NRC and JAF 54 EFPY Mean RT_{NDT} Calculations to the 64 EFPY Mean RT_{NDT} Calculations for the Limiting Combustion Engineering Owners Group Case Study on BWRVIP-05.

Table 1

Parameter Description	Limiting 64 EFPY CEOG Case Study	NRC 54 EFPY Mean RT_{NDT} Calculations for JAFNPP (1)	Applicant 54 EFPY Mean RT_{NDT} Calculations for JAF (1)
Alloy % Cu	0.183	0.337	0.337
Alloy % Ni	0.704	0.609	0.609
$RT_{NDT(U)}$ (°F)	0	-50	-50
Fluence (10^{19} n/cm^2,E>1.0 MeV)	0.4	0.253	0.253
Chemistry Factor	172.2	209.1	209.1
Δ RT_{NDT} (°F)	128.5	132.8	131.1
Mean RT_{NDT} (°F)	128.5	82.8	81.1
NRC Established Conditional Probability of Failure [P(F/E)] Criterion for Case / Result for Plant Specific Calculation	4.38×10^{-4} (Maximum P(F/E) value to justify relief)	Mean RT_{NDT} is lower than Case Study Mean RT_{NDT} : Criterion is met. (2)	Mean RT_{NDT} is lower than Case Study Mean RT_{NDT} : Criterion is met. (2)

(1) For the reactor vessel, the limiting circumferential weld materials determined by the staff were equivalent to those determined by the applicant. The limiting reactor vessel circumferential weld is 1-240 fabricated from weld heat No. 305414.

(2) If the plant-specific mean RT_{NDT} is less than the mean RT_{NDT} of the limiting case study, the staff concludes that probability of failure for the plant-specific circumferential weld under review will be less than the conditional probability of failure for the limiting circumferential weld in the limiting case study. BWR plants that meet this criterion may conclude that the probability of failure for the limiting circumferential reactor vessel welds is sufficiently low to justify elimination of both volumetric examinations required by ASME Code Section XI (Examination Category B-A, Item B1.11) and augmented volumetric examinations for the circumferential welds required by 10 CFR 50.55a(g)(6)(ii)(A)(2).

In JAF's response, dated February 12, 2007, to RAI 4.2.5-2, dated January 12, 2007, regarding confirmation on whether previous volumetric examinations of the reactor vessel axial shell welds showed any indication of cracking or other age-related degradation mechanisms, JAF stated that no unacceptable in-service examination indications have been found on reactor vessel welds, circumferential or axial. The staff found JAF's response to RAI 4.2.5-2 acceptable.

In JAF's UFSAR supplement summary description of its TLAA evaluation of reactor vessel circumferential weld inspection relief in LRA Section A.2.2.1.5 it states:

> The JAF reactor pressure vessel circumferential weld parameters at 54 EFPY will remain within the NRC's (64 EFPY) bounding CEOG parameters from the BWRVIP-05 SER. Although a conditional failure probability has not been calculated, the fact that the JAF values at the end of the license are less than the 64 EFPY value provided by the NRC leads to the conclusion that the JAF RPV conditional failure probability is bounded by the NRC analysis. As such, the conditional probability of failure for circumferential welds remains below that stated in the NRC's Final Safety Evaluation of BWRVIP-05. Therefore, this analysis has been projected through the period of extended operation per 10 CFR 54.21(c)(1)(ii).

The staff found JAF's UFSAR supplement summary description consistent with the staff analysis for the TLAA of the reactor vessel circumferential weld examination relief in SER Section 4.2.5.2. Therefore, based on this assessment, the staff found the UFSAR supplement summary description for the TLAA of the reactor vessel circumferential weld examination relief acceptable.

Based on the information presented in this request and the referenced LAR with the corresponding NRC SER, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds as stated in the staff's July 30, 1998 safety evaluation.

(2) Operator Training and Established Procedures that Limit the Frequency of Cold Over-Pressure Events

The procedures and training used to limit cold over-pressure events are the same as those approved by the NRC in the SER dated February 22, 2000 (Reference TAC No. MA6215, Accession No. ML003685801).

Review of Potential High Pressure Injection Sources:

The high-pressure make-up systems at FitzPatrick (i.e., the Feedwater, High Pressure Coolant Injection (HPCI), and the Reactor Core Isolation Cooling (RCIC) systems) are steam turbine driven. During reactor cold shutdown conditions, no steam is available for operation of these systems. Therefore, it is not plausible for these systems to contribute to an over-pressurization event while the unit is in cold shutdown. During reactor cold shutdown conditions the condensate booster pumps are normally maintained in the "pull-to-lock" position and the feedpump discharge isolation valves are normally maintained in the closed position. It would require several Operator errors and breakdowns in the work control process to inadvertently start a condensate booster pump and inject into the vessel. As discussed below, operating procedural restrictions, operator training, and work control processes at JAF provide appropriate controls to minimize the potential for RPV cold over-pressurization events.

During normal cold shutdown conditions, RPV level and pressure are controlled with the Control Rod Drive (CRD) and Reactor Water Cleanup (RWCU) systems using a "feed and bleed" process. The RPV is not taken solid during these items, and plant procedures require opening of the head vent valves after the reactor has been cooled to less than 212°F. If either of these systems were to fail, the Operators would adjust the other system to control level. Under these conditions, the CRD system typically injects water into the reactor at a rate of <60 gpm. This slow injection rate allows the operators sufficient time to react to unanticipated level changes and, thus, significantly reduces the possibility of an event that would result in a violation of the pressure-temperature limits.

The Standby Liquid Control (SLC) system is another high-pressure water source to the RPV. However, there are no automatic starts associated with this system. SLC injection requires Operators to manually start the system from the Control Room or from the local test station. Additionally, the injection rate of the SLC pump is approximately 50 gpm, which would give the Operators ample time to control reactor pressure in the case of an inadvertent injection.

Pressure testing of the RPV is classified as an "Infrequently Performed Test or Evolution" which ensures that these tests receive special management oversight and procedural controls to maintain the plant's level of safety within acceptable limits. The pressure test is conducted so that the required temperature bands for the pressure increases are achieved and maintained prior to increasing pressure. During performance of an RPV pressure test, level and pressure are controlled using the CRD and RWCU systems using a "feed and bleed" process. Increase in pressure is limited to less than 30 psig per minute. This practice, performed in accordance with site Surveillance Test (ST) procedures, minimizes the likelihood of exceeding the pressure-temperature limits during performance of the test.

Procedural Controls/Operator Training to Prevent Reactor Pressure Vessel Cold Over-Pressurization:

Operating procedural restrictions, operator training, and work control processes at JAF provide appropriate controls to minimize the potential for RPV cold over-pressurization events.

During normal cold shutdown conditions, reactor water level, pressure, and temperature are maintained within established bands in accordance with operating procedures. The

Operations procedure governing Control Room activities requires that Control Room Operators frequently monitor for indications and alarms to detect abnormalities as early as possible. This procedure also requires that the Shift Manager be notified immediately of any changes or abnormalities in indications. Furthermore, changes that could affect reactor level, pressure, or temperature can only be performed under the knowledge and direction of the Shift Manager or Control Room Supervisor. Therefore, any deviations in reactor water level or temperature from a specified band will be promptly identified and corrected. Finally, plant conditions and on-going activities that could affect critical plant parameters are discussed at each shift turnover. This ensures that on-coming Operators are cognizant of activities that could adversely affect reactor level, pressure, or temperature.

Procedural controls for reactor temperature, level, and pressure are an integral part of Operator training. Specifically, operators are trained in methods of controlling water level within specified limits, as well as responding to abnormal water level conditions outside the established limits. Additionally, Control Room Operators receive training on brittle fracture limits and compliance with the Technical Specification pressure-temperature limits curves. Plant-specific procedures have been developed to provide guidance to the Operators regarding compliance with the Technical Specification requirements on pressure-temperature limits.

During plant outages the work control processes ensure that the outage schedule and changes to the schedule receive a thorough shutdown risk assessment review to ensure defense-in-depth is maintained. At JAF outage work requests are scheduled by the Planning Scheduling and Outage department. Senior Reactor Operators assigned to the work control center provide oversight of outage schedule development to avoid conditions which could adversely impact reactor water level, pressure, or temperature. From the outage schedule, a daily schedule is developed listing the work activities to be performed. These daily schedules are reviewed and approved by Management, and a copy is maintained in the Control Room. Changes to the schedule require Management review and approval.

In the Control Room, the Shift Manager is required, by procedure, to maintain cognizance of any activity that could potentially affect reactor level or decay heat removal during refueling outages. The Control Room Operator is required to provide positive control of reactor water level and pressure within the specified bands, and promptly report when operating outside the specified band, including restoration actions being taken. Pre-job briefings are conducted for complex work activities, such as RPV pressure tests or hydrostatic testing that have the potential of affecting critical RPV parameters. Expected plant responses and contingency actions to address unexpected conditions, or responses that may be encountered, are included in the briefing discussion.

6. Duration of Proposed Alternative

The duration of this request is for the period of extended operation ending October 17, 2034.

In LRA RAI 4.2.5-1 dated January 12, 2007, the NRC asked JAF when it would apply for relief from ASME Code Section XI circumferential weld examination requirements for the period of extended operation. In the RAI response dated February 12, 2007, JAF stated that it will submit a request for relief from the circumferential weld examination requirements for each 10-year ISI interval in the period of extended operation.

7. Precedents

A similar request has been approved for the Peach Bottom Atomic Power Station, Units 1 and 2 (Reference Accession Number ML 112770217).

8. References:

1. NRC Generic Letter 98-05, “Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds,” dated November 10, 1998.
2. EPRI TR-105697, BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Inspection Recommendations (BWRVIP-05), September 1995.
3. NRC Letter from Gus C. Lainas, Acting Director, Division of Engineering, Office of Nuclear Regulatory Regulation, to Carl Terry, BWRVIP Chairman, Niagara Mohawk Company, “Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report,” dated July 28, 1998.
4. NRC Letter from Brian W. Sheron, Director, Division of Engineering, Office of Nuclear Reactor Regulation, to Carl Terry, BWRVIP Chairman, Niagara Mohawk Company, “transmittal of NRC Staff’s Independent Assessment of the Boiling Water Reactor Vessel and Internals Project BWRVIP-05 Report and Proprietary Request for Additional Information,” dated August 14, 1997.
5. BWRVIP Letter, Carl Terry, BWRVIP Chairman, Niagara Mohawk Company, to the NRC, C. E. Carpenter, “BWRVIP Response to NRC Request for Additional Information on BWRVIP-05,” dated December 18, 1997.
6. JAF-ICD-RPV-03393, BWRVIP Integrated Surveillance Program Data Verification Activity, Revision 0, dated July 26, 1999.
7. NYPA Letter (JPN-98-039), “Request for Additional Information Regarding Response to Generic Letter 92-01: Reactor Pressure Vessel Integrity (TAC No. MA1190), dated August 31, 1998.
8. NUREG-1905, U.S. NRC Safety Evaluation Report Related to the License Renewal of James A. FitzPatrick Nuclear Power Plant, Docket No. 50-333, dated April 2008.

9. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," Task SI 502-4, dated January 1987.