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-16-0137
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U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

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

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

Reference:
1) ENOI letter, 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, JAFP-15-0108, dated August 18, 2015
2) ENOI letter, Withdrawal of Proposed Alternative, Implementation of BWRVIP-05 (GL 98-05) at James A. FitzPatrick Nuclear Power Plant (JAF) (TAC No. MF6616), JAFP-16-0048, dated March 15, 2016

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 February 3, 2017. The ISI Code of Record for the fourth interval is ASME Section XI, 2001 Edition through the 2003 Addenda.

This request was originally submitted by Reference 1 and retracted by Reference 2; however, a refueling outage was recently re-scheduled in January 2017. JAF requests NRC Staff review and approval of this proposed alternative on or before January 9, 2017.

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) 349-6766.

Very truly yours,

William C. Drews
Regulatory Assurance Manager

WCD:mh

Attachment: James A. FitzPatrick Nuclear Power Plant In-service Inspection Program RR-19
Enclosure 1: Engineering Change (EC) 62614
Enclosure 2: Engineering Change (EC) 65540

cc: USNRC, Region I Administrator
USNRC, Project Manager
USNRC, Resident Inspector

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**Attachment**

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

**James A. FitzPatrick Nuclear Power Plant**
**In-service Inspection Program RR-19**

Proposed Alternative in Accordance with 10 CFR 50.5a(z)(1)
Implementation of BWRVIP-05

## 1. ASME Code Component(s) Affected

| | |
|---|---|
| Unit: | James A. FitzPatrick Nuclear Power Plant (JAF) |
| Inspection Interval: | Fourth ($4^{th}$) 10-year interval starting March 1, 2007, February 3, 2017 and the Period of Extended Operation October 17, 2034. |
| Code Class: | ASME Section XI Code Class 1 |
| Component Numbers: | VC-1-2, VC-2-3, VC-3-4, VC-4-BH-1 |

## 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

BWRVIP-05, “BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations” (Reference 1) evaluated that the failure frequency for circumferential welds in BWR plants is sufficiently low enough to be below the criterion specified in Regulatory Guide 1.154. The NRC endorsed this position with the staff’s evaluation of the BWRVIP-05 report issued July 28, 1998 (Reference 2). In 1998, in accordance with BWRVIP-05, JAF requested relief and gained approval (Reference 4) with Relief Request 17. This relief’s duration was evaluated for the initial Operating License Period ending on October 17, 2014.

JAF is requesting Relief Request 19 in order to apply BWRVIP-05 to the period of extended operation ending on October 17, 2034. This request is relevant to the current $4^{th}$ Inservice Inspection Interval because, without relief, inspection of circumferential shell welds are required in accordance with the applicable ASME Code requirements.

## 4. Proposed Alternative

The alternative plan would require performance of RPV vertical weld examinations and incidental examination of 2 to 3 percent of the intersecting circumferential shell welds to the maximum extent possible based on accessibility. The circumferential welds would be permanently deferred until plant renewed operating license expiration. This alternative aligns with BWRVIP-05.

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.

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## 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 1) 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 2). In this letter, the staff concluded that "since the failure frequencies for circumferential welds in BWR plants are significantly below the criteria specified in Regulatory Guide (RG) 1.154 and the CDF of any BWR plant, and that continued future inspections would result in a negligible decrease in an already acceptably low value, elimination of ISI for RPV circumferential welds is justified."

The staff's letter indicated that Boiling Water Reactor (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 (Reference 2) 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 (Reference 2) 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 License Renewal Application (LRA), and the applicant would need to 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).

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

During the review of the JAF LRA, the staff evaluated relief from the ASME Code Section XI circumferential weld examination requirements for the period of extended operation.

> Section 4.2.5.2 of the SER (Reference 3) 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. Based on this independent assessment, the staff concludes that the applicant's TLAA on circumferential weld relief requests conforms to Renewal Applicant Action No. 11 on Topical Report BWRVIP-74-A, has been projected to 54 EFPY, and is acceptable pursuant to 10 CFR 54.21(c)(1)(ii).

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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 JAFNPP 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 JAFNPP (1)** |
|---|---|---|---|
| Alloy % Cu | 0.183 | 0.337 | 0.337 |
| Alloy % Ni | 0.704 | 0.609 | 0.609 |
| $RT_{NDT(U)}$ (ºF) | 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 $RT_{NDT}$ (ºF) | 128.5 | 82.8 | 81.1 |
| NRC Established Conditional Probability of Failure [ P(F/E) ] Criterion for Case / Result for Plant Specific Calculation | $4.38 \times 10^{-4}$ (Maximum P(F/E) value to justify relief) (2) | 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 inservice 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 and is resolved.

In JAF's UFSAR supplement summary description of its Time-Limited Aging Analysis (TLAA) evaluation of reactor vessel circumferential weld inspection relief in LRA Section A.2.2.1.5 it states:

> The JAFNPP 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 JAFNPP values at the end of the license are less than the 64 EFPY value provided by

the NRC leads to the conclusion that the JAFNPP 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 LRA with the corresponding NRC SER (Reference 3), the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds as stated in the staff's July 28, 1998 safety evaluation (Reference 2).

**(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 similar as those approved by the NRC in their SER of JAF Relief Request RR-17 (Reference 4).

The staff also concluded the following in JAF SER 4.2.5.4 (Reference 3):

> "The staff reviewed the applicant's TLAA of the reactor vessel circumferential weld examination relief, as summarized in LRA Section 4.2.5, including RAI responses dated February 12, 2007. The staff determines that the applicant appropriately described how the conditional failure probability for the reactor vessel circumferential welds are bounded by the analysis in the staff SER dated July 28, 1998, on the BWRVIP-05 Report and how procedures and training will limit cold over-pressure events during the period of extended operation. The staff, therefore, concludes that the applicant's TLAA Section 4.2.5 and UFSAR supplement A.2.2.1.5 for reactor vessel circumferential weld examination relief will comply with the 10 CFR 54.21(c)(1)(ii) acceptance criterion for TLAAs."

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 feed pump 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 JAFNPP 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 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 JAFNPP 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 JAFNPP outage work requests are scheduled by the work control center. 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.

During outages, work is coordinated through the work control center, which provides an additional level of Operations oversight. 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. Cognizant individuals involved in the work activity attend pre-job briefings. Expected plant responses and contingency actions to address unexpected conditions, or responses that may be encountered, are included in the briefing discussion.

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 3). The one exception requiring further action by JAF affecting the Reactor Vessel Neutron Fluence TLAAs was license condition 4.2.1-1 which has been closed. Closure discussion is provided in Reference 3, section 1.5, page 1-7.

## 6. Lower Head Events

There were two violations of the Technical Specification Limiting Condition for Operation (TS LCO) 3.4.9 limits on Reactor Coolant System (RCS) heatup and cooldown rate in 2016.

The first event occurred on January 24, 2016 when a difference of 124.9 $^{o}$F over a 1-hour interval was noted on plant instrumentation. This exceeded the Technical Specification Requirements that RCS temperature change averaged over a one-hour period shall be $\leq$ 100 deg. F/hr. during heatup and cooldown operations. This was entered into the Corrective Action Program and the event was evaluated and found acceptable in Engineering Change (EC) 62614 (Reference 5). A copy of this EC is included in this submittal as Enclosure 1.

The second event occurred on June 25, 2016 when a difference of 125.7 $^{o}$F over a 1-hour interval was noted on plant instrumentation. This exceeded the Technical Specification Requirements that RCS temperature change averaged over a one-hour period shall be $\leq$ 100 deg. F/hr. during heatup and cooldown operations. This was entered into the Corrective Action Program and the event was evaluated and found acceptable in EC 65540 (Reference 6). A copy of this EC is included in this submittal as Enclosure 2.

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In both cases these events were evaluated for RPV thermal cycle limits and fatigue usage of the Control Rod Drive (CRD) nozzle stub tube and found to be acceptable. RPV bottom head temperature remained within the acceptable region of Figure 2 of the JAF Pressure-Temperature Limits Report (Technical Requirements Manual, Appendix F) throughout the heatup transient; therefore there is no brittle fracture concern associated with these events.

## 7. Duration of Proposed Alternative

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

## 8. Precedents

A similar request has been approved for the Peach Bottom Atomic Power Station, Units 1 and 2 (Reference Accession Number ML112770217).

## 9. References:

1. BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Inspection Recommendations (BWRVIP-05), EPRI TR-105697, September 1995
2. 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
3. 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
4. NRC letter, Relief Request No. 17 – Request for Relief from the Requirements of 10 CFR 50.55a(g)(6(ii)(A)(2) for augmented inspection of the Circumferential Welds in the Reactor Vessel of the James A. FitzPatrick Nuclear Power Plant (TAC No. MA6215), ML003685801, dated February 22, 2000
5. EC-JAF-62614, 1-24-16 RX Heatup Rate Exceedance Effect on RPV
6. EC-JAF-65540, 6-25-16 RX Heatup Rate Exceedance Effect on RPV

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**Enclosure 1**

**Engineering Change (EC) 62614**

CR-JAF-2016-00280 documents a violation of the Technical Specification Limiting Condition for Operation (TS LCO) 3.4.9 limits on Reactor Coolant System (RCS) heatup and cooldown rate. Specifically, Reactor Pressure Vessel (RPV) bottom head metal temperature obtained from EPIC-A-1020 documented during performance of ST-26J, "Heatup and Cooldown Temperature Checks" increased from 102.8°F at 1-24-16 22:15 to 211.9°F at 1-24-16 23:15, a difference of 109.1°F over an one hour interval. Review of PI data for this EPIC point over the same interval show that RPV bottom head temperature had a minimum value of 99.2°F with a maximum value of 224.1°F for a difference of 124.9°F. This violated the TS Surveillance Requirement (TS SR) 3.4.9.1.b.3 limit that RCS temperature change averaged over a one hour period be ≤ 100°F/hr during heatup and cooldown operations. The purpose of the LCO limits are to ensure stress limits for cyclic operation are not exceeded such that integrity of the Reactor Coolant Pressure Boundary (RCPB) is maintained (TS Bases B 3.4.9).

A previous violation of RPV cooldown rate (DER-95-1302) was evaluated in General Electric letter report GENE-523-A087-0995, dated 9/8/95. GENE-523-A087-0995 determined that a vessel bottom head heatup rate of 200°F/hr is acceptable with respect to fracture toughness on the basis of a cooldown rate of the same magnitude being acceptable and RPV heatup producing compressive stresses in the RPV wall versus the tensile stresses associated with cooldown. This conclusion was based on comparison of the JAF RPV configuration with a BWR/6 vessel for which detailed evaluation had been performed. While GENE-523-A087-0995 was written 20 years ago, the RPV bottom head is not subject to significant neutron fluence so the conclusions of the report remain valid.

GENE-523-A087-0995 states that the stress cycle associated with RPV bottom head cooldown and subsequent heatup is bounded by the thermal cycle associated with a Loss of Feedwater Pump transient in the RPV stress analysis. The latest RPV Thermal Cycles report (RAP-7.4.10-151231-52640268) shows that the RPV has undergone 5 of these cycles (reference event 11) versus an allowable value of 12 cycles for the life of the plant (UFSAR Table 4.2-3). This provides significant margin with respect to cyclic fatigue for the RPV bottom head. That there is substantial margin with respect to cyclic fatigue for the RPV bottom head is also supported by the projection of DRN-03-00794 to SIR-02-045 that shows 60-year fatigue usage for the CRD nozzle stub tube to RPV junction of 0.0010 and housing to stub tube junction of 0.0234.

RPV bottom head temperature remained within the acceptable region of Figure 2 of the JAF Pressure-Temperature Limits Report (Technical Requirements Manual, Appendix F) throughout the heatup transient, therefore there are no brittle fracture concerns associated with this event.

Note that this event is similar to another previous event at JAF evaluated in CR-JAF-2003-03925, CA 1.

<u>References</u>

1) CR-JAF-2016-00280

2) JAFNPP Technical Specifications

3) ST-26J, "Heatup and Cooldown Temperature Checks," Rev. 23

4) DER-95-1302, "During Post Scram Cool Down the Vessel Temp & Press Went to the Left Side of the P-T Curve" (see P2E 10.003)

5) GE Letter Report, GENE-523-A087-0995, "FitzPatrick Vessel Bottomhead Pressure-Temperature Transient Event," September 8, 1995 (contained in either P2E 10.003 or 10.004)

6) RAP-7.4.10-151231-52640268, "Component Cyclic or Transient Limit Program Period 7-1-15 Thru 12-31-15 WO-52640268-01"

7) JAFNPP Updated Final Safety Analysis Report (UFSAR)

8) DRN-03-00794, "Updated Fatigue Analysis for JAF RPV Components MCC #1A

9) SIR-02-045, "Updated Fatigue Analysis for JAF – Reactor Pressure Vessel Components"

10) JAFNPP Technical Requirements Manual

11) CR-JAF-2003-03925 (See P2E 10.004)

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**Enclosure 2**

**Engineering Change (EC) 65540**

CR-JAF-2016-02435 documents a violation of the Technical Specification Limiting Condition for Operation (TS LCO) 3.4.9 limits on Reactor Coolant System (RCS) heatup and cooldown rate. Specifically, Reactor Pressure Vessel (RPV) bottom head metal temperature obtained from EPIC-A-1020 documented during performance of ST-26J, "Heatup and Cooldown Temperature Checks" increased from 105.9°F at 6-25-16 22:30 to 230.9°F at 6-25-16 22:45, a difference of 125°F within a one hour interval. Review of PDS data for this EPIC point over the same interval show that RPV bottom head temperature had a minimum value of 105.6°F with a maximum value of 231.3°F for a difference of 125.7°F. This violated the TS Surveillance Requirement (TS SR) 3.4.9.1.b.3 limit that RCS temperature change averaged over a one hour period be ≤ 100°F/hr during heatup and cooldown operations. The purpose of the LCO limits are to ensure stress limits for cyclic operation are not exceeded such that integrity of the Reactor Coolant Pressure Boundary (RCPB) is maintained (TS Bases B 3.4.9).

EPIC-A-1020 receives a signal from 02TE-106, "ADS REACTOR VESSEL BOTTOM DRAIN TEMP ELEMENT" and measures temperature at the Reactor Pressure Vessel (RPV) bottom head drain line (which is indicative of the metal temperature on the inside wall of the lower RPV bottom head). Temperature measurements from 02-3TE-69L1, "NBI REACTOR VESSEL BOTTOM HEAD TEMP ELEMENT" as obtained from 02-3TR-89, "REACTOR VESSEL TEMP RECORDER," point 11 were also recorded in ST-26J for this time interval. The maximum change in RPV metal temperature measured at the outside surface of the lower RPV bottom head by 02-3TE-69L1 over a one hour interval was from 117°F at 6-25-16 22:30 to 181°F at 6-25-16 23:30. This value is within the 100°F/hr heatup rate required by Technical Specifications.

A previous violation of RPV cooldown rate (DER-95-1302) was evaluated in General Electric letter report GENE-523-A087-0995, dated 9/8/95. GENE-523-A087-0995 determined that a vessel bottom head heatup rate of 200°F/hr is acceptable with respect to fracture toughness on the basis of a cooldown rate of the same magnitude being acceptable and RPV heatup producing compressive stresses in the RPV wall versus the tensile stresses associated with cooldown. This conclusion was based on comparison of the JAF RPV configuration with a BWR/6 vessel for which detailed evaluation had been performed. While GENE-523-A087-0995 was written 20 years ago, the RPV bottom head is not subject to significant neutron fluence so the conclusions of the report remain valid.

GENE-523-A087-0995 states that the stress cycle associated with RPV bottom head cooldown and subsequent heatup is bounded by the thermal cycle associated with a Loss of Feedwater Pump transient in the RPV stress analysis. The latest RPV Thermal Cycles report (RAP-7.4.10-151231-52640268) shows that the RPV has undergone 5 of these cycles (reference event 11) versus an allowable value of 12 cycles for the life of the plant (UFSAR Table 4.2-3). This provides significant margin with respect to cyclic fatigue for the RPV bottom head (even considering the January 2016 transient in which the heatup rate was violated for similar causes to this event, CR-JAF-2016-00280). That there is substantial margin with respect to cyclic fatigue for the RPV bottom head is also supported by the projection of DRN-03-00794 to SIR-02-045 that shows 60-year fatigue usage for the CRD nozzle stub tube to RPV junction of 0.0010 and housing to stub tube junction of 0.0234.

RPV bottom head temperature remained within the acceptable region of Figure 2 of the JAF Pressure-Temperature Limits Report (Technical Requirements Manual, Appendix F) throughout the heatup transient, therefore there are no brittle fracture concerns associated with this event.

Note that this event is similar to previous events at JAF evaluated in CR-JAF-2003-03925, CA 1 and EC-62614 (CR-JAF-2016-00280).

References

1) CR-JAF-2016-02435

2) EC-62614

3) CR-JAF-2016-00280

4) JAFNPP Technical Specifications

5) ST-26J, "Heatup and Cooldown Temperature Checks," Rev. 23

6) DER-95-1302, "During Post Scram Cool Down the Vessel Temp & Press Went to the Left Side of the P-T Curve" (see EC-62614 P2E 10.003)

7) GE Letter Report, GENE-523-A087-0995, "FitzPatrick Vessel Bottomhead Pressure-Temperature Transient Event," September 8, 1995 (contained in either EC-62614 P2E 10.003 or 10.004)

8) RAP-7.4.10-151231-52640268, "Component Cyclic or Transient Limit Program Period 7-1-15 Thru 12-31-15 WO-52640268-01"

9) JAFNPP Updated Final Safety Analysis Report (UFSAR)

10) DRN-03-00794, "Updated Fatigue Analysis for JAF RPV Components MCC #1A

11) SIR-02-045, "Updated Fatigue Analysis for JAF – Reactor Pressure Vessel Components"

12) JAFNPP Technical Requirements Manual

13) CR-JAF-2003-03925 (See EC-62614 P2E 10.004)