Exelon Generation

James A. FitzPatrick NPP
P.O. Box 110
Lycoming, NY 13093

William C. Drews
Regulatory Assurance Manager

JAFP-18-0113
December 17, 2018

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

James A. FitzPatrick Nuclear Power Plant
Renewed Facility Operating License No. DPR-59
NRC Docket No. 50-333

Subject: Fifth Ten-Year Inservice Inspection Interval Program Plan

Dear Sir or Madam:

The Fifth 10-Year Inservice Inspection (ISI) Interval began at James A. FitzPatrick Nuclear Power Plant on August 1, 2017. The Enclosure to this letter contains the ISI program plan for your records; submitted in accordance with IWA-1400(c) of the 2007 Edition with the 2008 Addenda of American Society of Mechanical Engineers (ASME) Section XI.

There are no new regulatory commitments contained in this letter.

If you have any questions concerning, please contact William Drews, Regulatory Assurance Manager, at (315) 349-6562.

Sincerely,

William C. Drews
Regulatory Assurance Manager

WD/mh

Enclosure: ER-JF-330-1001 ISI Program Plan Fifth Ten-Year Inservice Inspection Interval

cc:
NRC Regional Administrator, Region I
NRC Resident Inspector
NRC Project Manager

JAFP-18-0113
Enclosure

ER-JF-330-1001 ISI Program Plan Fifth Ten-Year Inservice Inspection Interval

(163 Pages)

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

ISI PROGRAM PLAN
FIFTH TEN-YEAR INSERVICE INSPECTION INTERVAL

Commercial Service Date:

7/28/1975

James A. FitzPatrick Nuclear Power Plant
268 Lake Road
Oswego, NY 13126

Exelon Generation Company, LLC (EGC)
200 Exelon Way
Kennett Square, PA 19348

REVISION APPROVAL SHEET

TITLE: ISI and CISI Program Plan
Fifth Ten-Year Inservice Inspection Interval
James A. FitzPatrick Nuclear Power Plant

ASME CODE OF RECORD: ASME Section XI, 2007 Edition with the 2008 Addenda

DOCUMENT: ER-JF-330-1001 REVISION: 1

Prepared: Chelsea Gill, 11/8/18
Chelsea Gill
Site Material Programs Engineer

Reviewed: Brendan J. Casey, 11/8/2018
Brendan Casey
Fleet ISI Engineer

Approved: , 11/08/18
Keith Schoales
JAF Engineering Programs Manager

Reviewed: , 11/08/18
Brandon Powers
Authorized Nuclear Inservice Inspector (ANII)

Each time this document is revised, the Revision Approval Sheet will be signed and the following Revision Control Sheet should be completed to provide a detailed record of the revision history. The signatures above apply only to the changes made in the revision noted. If historical signatures are required, JAF archives should be retrieved.

REVISION CONTROL SHEET

Major changes to this document should be outlined within the table below. Editorial and formatting revisions are not required to be logged.

Revision	Date	Revision Summary
1	11/8/2018	Updated Table 8.0-1 for recent approvals and withdrawals. Updated Table 5.1-1 in Section 5.0. Deleted Section 5.2 Roles and Responsibilities from Section 5.0. Incorporated ATI 04088160-07 to include IWE acceptance criteria. Removed repetitive figures of Figure 1 in Section 2.2.1. Added references for IWE examinations of the liner interface. Incorporated ATI 04186269-02 to describe the containment monitoring commitment and 04186270-02 to update section 2.3. Updated Section 9.0. Updated Table 7.1-1. Incorporate requirements of NER NC-18-005-Y.
0	7/30/2018	Converted procedure to T&RM from ISI-JAF-LTP5-PLAN to ER-JF-330-1001. Updated to incorporate status of relief requests. Added code cases N-702 and N-513-4, removed code cases N-416-4. Updated to include 10CFR50.55a and Regulatory Guide 1.147 revisions. Added Pressure Testing to plan in order to delete procedure SEP-PT-JAF-001. Clarified IGSCC timeline. Corrected various grammatical errors.

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1.0 INTRODUCTION AND BACKGROUND

1.1 Introduction

This Inservice Inspection (ISI) Program Plan details the requirements for the examination and testing of ISI Class 1, 2, 3, and MC pressure retaining components, supports, and containment structures at James A. FitzPatrick Nuclear Power Plant (JAF). This ISI Program Plan also includes Containment Inservice Inspection (CISI), Risk-Informed Inservice Inspection (RI-ISI), Augmented Inservice Inspection (AUG), and System Pressure Testing (PT) requirements imposed on or committed to by JAF.

The ISI Program Plan is also credited for License Renewal. JAF's current (original) operating license (DPR-59) expired on October 17, 2014. Specific requirements must be fulfilled for completion of the renewal application. Program requirements based on the license renewal commitments are detailed in their respective subsections in this document. Below is a list of items that fall within the scope of the license renewal.

- BWR Feedwater Nozzle
  - LRA Appendix B.1.3 (P-18369 LO-LAR-2008-0048 CA-32)
    - Continue ISI inspections of FW nozzle in accordance with ASME Section XI, Subsection IWB and GE NE-523-A71-0594. (JAF-RPT-09-LR003, dated April 18, 2011).
- Containment Inservice Inspection
  - LRA Appendix B.1.16.1 (P-18378 LO-LAR-2008-0048 CA-41)
    - This is a site specific generic commitment to maintain the CISI program as described in LRA Appendix B. (JAF-RPT-09-LR161, dated April 11, 2011).
- Inservice Inspection Program
  - LRA Appendix B.1.16.2 (P-18379 LO-LAR-2008-0048 CA-56)
    - Update documents as described in JAF report JAD-RPT-09-LR-162 in order to maintain the Inservice Inspection Program as described in LRA Appendix B, Section B1.16.2 and the AMPER (JAF-RPT-05-LRD02).
- BWR Stress Corrosion Cracking
  - LRA Appendix B.1.5 (P-18371 LO-LAR-2008-0048 CA-34)
    - ISI inspections required by the BWRVIP are completed by JAF by qualified personnel.
    - ISI sample expansion occurs consistent with BWRVIP-75-A guidelines (JAF-RPT-09-LR005, dated April 12, 2011).
- CRD Return Line Nozzle
  - LRA Appendix B.1.2 (P-18343 LO-LAR-2008-0048 CA-3,50)
    - Continue UT examinations of the CRD Return line nozzle to cap weld.
    - Continue EVT-1 visual examinations of the CRD return line nozzle blend radius and adjacent vessel wall.
    - Enhance the BWR CRD Return Line Nozzle Program to examine the CRDRL nozzle-to-vessel weld and the CRDRL nozzle inside radius section per Section XI Table IWB-2500-1, Category B-D Items B3.90 and B3.100. (JAFP-06-0109, dated July 31, 2006)
- BWR Penetrations Program
  - LRA Appendix B.1.4 (P-18370 LO-LAR-2008-0048 CA-33)

    - Continue core nozzle inspections in accordance with BWRVIP-27-A and instrument nozzle inspection in accordance with BWRVIP-49-A (JAF-RPT-09-LR004, dated April 18, 2011).
- Reactor Head Closure Studs
  - LRA Appendix B.1.23 (P-18380 LO-LAR-2008-0048 CA-43)
    - Maintain the reactor head closure studs program described in Appendix B. (JAF-RPT-09-LR023, dated May 18, 2011).
- CASS Embrittlement Program
  - LRA Appendix B.1.28 (A-18359 LO-LAR-2008-0048 CA-18)
    - Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in LRA Appendix B. (JAF-RPT-09-LR028, dated December 6, 2010).
- BWR Vessel ID Attachment Welds Program
  - LRA Appendix B.1.6 (P-18372 LO-LAR-2008-0048 CA-35)
    - Implement the Vessel ID Attachment Welds Program as described in LRA Appendix B (JAF-RPT-09-LR006, dated April 20, 2011).
- Lubrite Sliding Supports
  - LRA Appendix B.1.16.2 (P-18379 LO-LAR-2008-0048 CA-42)
    - Provide periodic inspections to confirm the absence of aging effects for lubrite sliding supports used in the torus supports.

At JAF, the Inservice Testing (IST) Program is maintained and implemented separately from the ISI Program. The IST Program Plan contains all applicable inservice testing requirements.

The Fifth ISI Interval for JAF is effective from August 1, 2017 through June 15, 2027. With the update to the ISI Program for the Fifth ISI Interval for ISI Class 1, 2, and 3 components, including their supports, Exelon Generating Company (Exelon) has also elected to update the CISI Program to its Third CISI Interval for ISI Class MC Components at the same time. During the Second Ten-Year CISI Interval, the CISI Program was aligned with the ISI Interval which enabled all of the ISI and CISI Program components / piping structural elements to be based on the same effective Edition and Addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, as well as share a common interval start and end date. The common ASME Code of Record for the Fifth ISI Interval and the Third CISI Interval is the 2007 Edition through the 2008 Addenda. The ISI and CISI Program Plans are controlled and revised in accordance with the requirements of procedure ER-AA-330, "Conduct of Inservice Inspection Activities," which implements the ASME Section XI ISI Program.

Paragraph IWA-2430(c)(1) of ASME Section XI allows an inspection interval to be reduced or extended by as much as one year, and Paragraph IWA-2430(d) allows an inspection interval to be extended for a period equivalent to the outage when a unit is out of service continuously for six months or more. Reference Tables 1.1-1 and 1.1-2 for intervals, periods, and extensions that apply to JAF's Fifth ISI Interval and Third CISI Interval.

The Fifth ISI Interval and the Third CISI Interval are divided into three inspection periods as determined by calendar years within the intervals. Tables 1.1-1 and 1.1-2 identify the period start and end dates for the Fifth ISI Interval and the Third CISI Interval as defined by Inspection Program IWA-2431. In accordance with Paragraph IWA-2430(c)(3), the inspection periods specified in these Tables may be reduced or extended by as much as 1 year.

TABLE 1.1-1
FIFTH ISI INTERVAL/PERIOD/OUTAGE MATRIX
(for ISI Class 1, 2, and 3 Components and Supports)

Interval	**Periods**	**Outages**	
Start Date to End Date	**Start Date to End Date**	**Outage Dates and/or Durations**	**Outage Numbers**
Fifth ISI Interval 08/01/2017 to 06/15/2027	1st 08/01/2017 to 12/31/2020	Scheduled Fall 2018	R23
		Scheduled Fall 2020	R24
	2nd 01/01/2021 to 12/31/2023	Scheduled Fall 2022	R25
	3rd 01/01/2024 to 06/15/2027	Scheduled Fall 2024	R26
		Scheduled Fall 2026	R27

TABLE 1.1-2
THIRD CISI INTERVAL/PERIOD/OUTAGE MATRIX
(for ISI Class MC Components and Supports)

<table>
<tr><th>Interval</th><th>Periods</th><th colspan="2">Outages</th></tr>
<tr><th>Start Date
to
End Date</th><th>Start Date
to
End Date</th><th>Outage
Dates and/or
Durations</th><th>Outage
Numbers</th></tr>
<tr><td rowspan="5"><u>Third CISI Interval</u>
08/01/2017
to
06/15/2027</td><td rowspan="2">1st
08/01/2017 to 12/31/2020</td><td>Scheduled
Fall 2018</td><td>R23</td></tr>
<tr><td>Scheduled
Fall 2020</td><td>R24</td></tr>
<tr><td>2nd
01/01/2021 to 12/31/2023</td><td>Scheduled
Fall 2022</td><td>R25</td></tr>
<tr><td rowspan="2">3rd
01/01/2024 to 06/15/2027</td><td>Scheduled
Fall 2024</td><td>R26</td></tr>
<tr><td>Scheduled
Fall 2026</td><td>R27</td></tr>
</table>

1.2 Background

The New York Power Authority (NYPA) obtained construction permit CPPR-71 to build JAF on May 20, 1970. The docket number assigned to JAF is 50-333. After satisfactory plant construction and preoperational testing was completed, JAF was granted a full power operating license, DPR-59. The commercial operating license date for JAF was July 28, 1975.

At the time JAF was constructed, the ASME Boiler and Pressure Vessel Code only covered nuclear vessels and associated piping up to and including the first isolation or check valve. Therefore, JAF's piping systems and associated components were designed and fabricated to the rules of USAS B31.1.0-1967 Edition. 10 CFR 50.55a(c)4 states that the applicable Code edition for a reactor coolant pressure boundary (RCPB) component is the Code Edition and Addenda approved by the NRC at the time of construction permits issuance (B31.1). The design and fabrication code for the JAF BWR Mark I containment is ASME Section III 1968 Edition including the 1968 Summer Addenda. The containment vessel is a Class "B" vessel as defined in this Code. The Reactor Vessel Code of Construction is the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition including the 1966 Winter Addenda.

1.3 First Interval ISI Program

On July 28, 1975, JAF began commercial operation, which marked the beginning of the First ISI Interval. Inservice Inspection and Repair and Replacement Programs were developed to implement the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI. The First ISI Interval for JAF ended on July 27, 1985. Since JAF was built to earlier editions of the ASME Code, the inservice inspection code in effect for the first two periods was ASME Section XI, 1970 Edition. The third inservice inspection period (Spring 1980) was conducted to the updated inservice inspection program in accordance with the 1974 Edition, 1975 Addenda of the ASME Code.

1.4 Second Interval ISI Program

Pursuant to the Code of Federal Regulations, Title 10, Part 50, Section 55a, *Codes and standards*, (10 CFR 50.55a), Paragraph (g), *Inservice inspection requirements*, licensees were required to update their ISI Programs at the end of the First ISI Interval. The ISI Program was required to comply with the latest Edition and Addenda of the Code incorporated by reference in 10 CFR 50.55a twelve months prior to the start of the interval per 10 CFR 50.55a(g)(4)(ii).

The Second ISI Interval commenced on July 28, 1985. As allowed by ASME Section XI, Paragraph IWA-2430(d), the Second Inspection Interval was extended to September 27, 1997 to account for the extended period of outage time during the Second Interval.

Therefore, the JAF's Second ISI Interval was effective from July 28, 1985 through September 27, 1997.

1.5 Third Interval ISI Program

Pursuant to 10 CFR 50.55a(g), licensees are required to update their ISI Programs to meet the requirements of ASME Section XI once every ten years or inspection interval. The ISI Program is required to comply with the latest Edition and Addenda of the Code incorporated by reference in 10 CFR 50.55a twelve months prior to the start of the interval per 10 CFR 50.55a(g)(4)(ii). The code edition used for the Third Interval ISI Program was ASME Section XI, 1989 Edition with no addenda. As discussed in Section 1.4 above, the start of the Third ISI Interval was on September 28, 1997 for JAF.

JAF's Third ISI Interval was effective from September 28, 1997 through February 28, 2007.

1.6 Fourth Interval ISI Program

JAF's Fourth Ten-Year Inspection Interval was effective March 1, 2007 through and including July 31, 2017. The ISI Program Plan was developed in accordance with the 2001 Edition through the 2003 Addenda. The operating license ended on October 17, 2014 and JAF entered the period of extended operation. It should be noted that the plant was scheduled to shutdown permanently until the sale of the plant to Exelon. Subsequently the NRC approved the license transfer to Exelon which became effective March 31, 2017. Due to this change in owner and an extended outage, the Fourth Ten-Year Inspection Interval was extended to and including July 31, 2017.

1.7 Fifth Interval ISI Program

Pursuant to 10CFR50.55a(g), licensees are required to update their ISI Programs to meet the requirements of ASME Section XI once every ten years or inspection interval. The ISI Program is required to comply with the latest Edition and Addenda of ASME Section XI incorporated by reference in 10 CFR 50.55a twelve months prior to the start of the interval per 10 CFR 50.55a(g)(4)(ii). As discussed in Section 1.6 above, the start of the Fifth ISI Interval for JAF will be on August 1, 2017. Based on this date, the latest Edition and Addenda of ASME Section XI referenced in 10 CFR 50.55a(b)(2) twelve months prior to the start of the Fifth ISI Interval was the 2007 Edition through the 2008 Addenda.

The JAF Fifth Interval ISI Program Plan was developed in accordance with the requirements of 10 CFR 50.55a including all published changes through December 11, 2014 and the 2007 Edition with the 2008 Addenda of ASME Section XI, subject to the conditions contained within Paragraph (b) of the regulation. These conditions are detailed in Table 1.11-1 of this section. Any revisions to 10 CFR 50.55a will be incorporated into this table. Fifth Interval ISI Program Plan addresses Subsections IWA, IWB, IWC, IWD, IWF, Mandatory Appendices, approved ASME Code Cases, and approved alternatives through relief requests and SER's.

The JAF Fifth ISI Interval is effective from August 1, 2017 through June 15, 2027.

JAF adopted Code Case N-716-1 for implementing risk-informed inservice inspections during the Fifth ISI Interval. Implementation of the RI-ISI Program is in accordance with RG 1.147 Rev. 17 which endorsed Code Case N-716-1.

1.8 First Interval CISI Program

CISI examinations were originally invoked by amended regulations contained within a Final Rule issued by the USNRC. The amended regulation incorporated the requirements of the 1992 Edition through the 1992 Addenda of the ASME Section XI, Subsection IWE, subject to specific modifications that were included in Paragraphs 10 CFR 50.55a(b)(2)(ix) and 10 CFR 50.55a(b)(2)(x).

The final rulemaking was published in the Federal Register on August 8, 1996 and specified an effective date of September 9, 1996. Implementation of the Subsection IWE Program from a scheduling standpoint was driven by the five year expedited implementation period per 10 CFR 50.55a(g)(6)(ii)(B), which specified that the examinations required to be completed by the end of the First Period of the First CISI Interval (per Table IWE-2412-1) be completed by the effective date (by September 9, 2001). For the second and third periods of the first Ten-Year CISI Inspection Interval, from March 2001 to September 2006, the code of record used was the 1998 Edition of ASME Section XI. NRC approval was made for this alternative on May 1, 2002 (TAC NO. MB2946).

The JAF First CISI Interval was effective from September 9, 1997 through September 9, 2006. Containment inservice examinations scheduled for the first 40-month period were completed during the Third Period of the Third ISI Inspection Interval. These examinations now serve the same purpose as preservice baseline examinations.

1.9 Second Interval CISI Program

The JAF Second Interval CISI Program Plan commenced on March 1, 2007 coincident with the start of the Fourth Ten-Year ISI Program Interval. The 2001 Edition with the 2003 Addenda of ASME Boiler and Pressure Vessel Code Section XI was used as the Code of Record for the Second Ten-Year CISI Interval.

Due to the change in owner and an extended outage, the Fourth ISI Ten-Year Inspection Interval was extended to and including July 31, 2017.

The JAF Second CISI Interval thus was effective from March 1, 2007 through and including July 31, 2017.

1.10 Third Interval CISI Program

Pursuant to 10 CFR 50.55a(g), licensees are required to update their CISI Programs to meet the requirements of ASME Section XI once every ten years or inspection interval. The CISI Program is required to comply with the latest Edition and Addenda of the Code incorporated by reference in 10 CFR 50.55a twelve months prior to the start of the interval per 10 CFR 50.55a(g)(4)(ii). As discussed in Section 1.9 above, the start of the Third CISI Interval will commence on August 1, 2017 for JAF. Based on this date, the latest Edition and Addenda of the Code referenced in 10 CFR 50.55a(b)(2) twelve months prior to the start of the Third CISI Interval was the 2007 Edition through the 2008 Addenda.

The JAF Third Interval CISI Program Plan was developed in accordance with the requirements of 10 CFR 50.55a including all published changes through December 11, 2014 and the 2007 Edition with the 2008 Addenda of ASME Section XI, subject to the conditions contained within Paragraph (b) of the regulation. These conditions are detailed in Table 1.11-1 of this section. This Third Interval CISI Program Plan addresses Subsection IWE, Mandatory Appendices, approved Code Cases, approved alternatives through relief requests and SER's, and utilizes the Inspection Program as defined therein.

1.11 Code of Federal Regulations 10 CFR 50.55a Requirements

There are certain paragraphs in 10 CFR 50.55a that list the conditions to the implementation requirements of ASME Section XI. These paragraphs in 10 CFR 50.55a that are applicable to the JAF scheduled ISI and CISI examination programs are detailed in Table 1.11-1.

TABLE 1.11-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a REQUIREMENTS

10 CFR 50.55a Paragraphs	Conditions
10 CFR 50.55a(b)(2)(ix)(A)	**(CISI)** *Metal containment examinations: First provision* For Class MC applications, the following apply to inaccessible areas: 1) N/A for the 2007 Edition through the 2008 Addenda 2) For each inaccessible area identified for evaluation, the applicant or licensee must provide the following in the ISI Summary Report as required by IWA-6000: (*i*) A description of the type and estimated extent of degradation, and the conditions that led to the degradation; (*ii*) An evaluation of each area, and the result of the evaluation, and; (*iii*) A description of necessary corrective actions.
10 CFR 50.55a(b)(2)(ix)(B)	**(CISI)** *Metal containment examinations: Second provision* When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.
10 CFR 50.55a(b)(2)(ix)(J)	**(CISI)** *Metal containment examinations: Tenth provision* In general, a repair/replacement activity such as replacing a large containment penetration, cutting a large construction opening in the containment pressure boundary to replace steam generators, reactor vessel heads, pressurizers, or other major equipment; or other similar modification is considered a major containment modification. When applying IWE-5000 to Class MC pressure-retaining components, any major containment modification or repair/replacement must be followed by a Type A test to provide assurance of both containment structural integrity and leak-tight integrity prior to returning to service, in accordance with 10 CFR

TABLE 1.11-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a REQUIREMENTS

10 CFR 50.55a Paragraphs	Conditions
	part 50 Appendix J, Option A or Option B on which the applicant's or licensee's Containment Leak-Rate Testing Program is based. When applying IWE-5000, if a Type A, B, or C Test is performed, the test pressure and acceptance standard for the test must be in accordance with 10 CFR part 50, Appendix J.
10 CFR 50.55a(b)(2)(x)	**(CISI) (ISI)** *Section XI condition: Quality assurance*. When applying Section XI editions and addenda later than the 1989 Edition, the requirements of NQA–1, "Quality Assurance Requirements for Nuclear Facilities," 1979 Addenda through the 1989 Edition of ASME BPV Code, Section XI, the edition and addenda of NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," 1994 Edition, the 2008 Edition, and the 2009-1a Addenda specified in IWA–1400 or Table IWA-1600-1 of that edition and addenda of Section XI, may be used by the licensee provided that the licensee uses it's appendix B to this part quality assurance program in conjunction with Section XI requirements and the commitments contained in the licensee's quality assurance program description. Where NQA-1 and Section XI do not address the commitments contained in the licensee's appendix B quality assurance program description, those licensee commitments must be applied to Section XI activities.
10 CFR 50.55a(b)(2)(xviii)(A)	**(CISI) (ISI)** *Section XI condition: NDE personnel certification. (A) NDE personnel certification: First provision.* Level I and II nondestructive examination personnel must be recertified on a 3-year interval in lieu of the 5 year interval specified in the 1997 Addenda and 1998 Edition of IWA-2314, and IWA-2314(a) and IWA-2314(b) of the 1999 Addenda through the latest edition and addenda incorporated by referenced in paragraph (a)(1)(ii) of this section.
10 CFR 50.55a(b)(2)(xix)	**(ISI)** *Section XI condition: Substitution of alternative methods.* The provisions for substituting alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA-2240 must be applied. The provisions in IWA-4520(b)(2) and IWA-4521 of the 2008 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, allowing the substitution of ultrasonic examination for radiographic examination specified in the Construction Code, are not approved for use.
10 CFR 50.55a(b)(2)(xxii)	**(ISI)** *Section XI condition*: *Surface Examination:* The use of the provision in IWA-2220, "Surface Examination," of Section XI, 2001 Edition through the latest Edition and Addenda incorporated

TABLE 1.11-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a REQUIREMENTS

10 CFR 50.55a Paragraphs	Conditions
	by reference in paragraph (a)(1)(ii) of this section, that allow use of an ultrasonic examination method is prohibited.
10 CFR 50.55a(b)(2)(xxvi)	**(ISI)** *Section XI condition: Pressure Testing Class 1, 2, and 3 Mechanical Joints.* The repair and replacement activity provisions in IWA-4540(c) of the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints must be applied when using the 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section.
10 CFR 50.55a(b)(2)(xxvii)	**(ISI)** *Section XI condition: Removal of Insulation.* When performing visual examinations in accordance with IWA-5242 of Section XI of the ASME BPV Code, 2003 Addenda through the 2006 Addenda or IWA-5241 of the 2007 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, insulation must be removed from 17-4 PH or 410 stainless steel studs or bolts aged at a temperature below 1100 °F or having a Rockwell Method C hardness value above 30, and from A-286 stainless steel studs or bolts preloaded to 100,000 pounds per square inch or higher.
10 CFR 50.55a(b)(2)(xxviii)	**(ISI)** *Section XI condition: Analysis of flaws.* Licensees using ASME BPV Code Section XI, Appendix A, must use the following conditions when implementing Equation (2) in A-4300(b)(1): For $R < 0$, ΔK_I depends on the crack depth (a), and the flow stress (σ_f). The flow stress is defined by $\sigma_f = 1/2(\sigma_{ys} + \sigma_{ult})$, where σ_{ys} is the yield strength and σ_{ult} is the ultimate tensile strength in units ksi (MPa) and (a) is in units in. (mm). For $-2 \leq R \leq 0$ and $K_{max} - K_{min} \leq 0.8 \times 1.12\ \sigma_f \sqrt{(\pi a)}$, $S = 1$ and $\Delta K_I = K_{max}$. For $R < -2$ and $K_{max} - K_{min} \leq 0.8 \times 1.12\ \sigma_f \sqrt{(\pi a)}$, $S = 1$ and $\Delta K_I = (1 - R)\ K_{max}/3$. For $R < 0$ and $K_{max} - K_{min} > 0.8 \times 1.12\ \sigma_f \sqrt{(\pi a)}$, $S = 1$ and $\Delta K_I = K_{max} - K_{min}$.
10 CFR 50.55a(b)(5)	**(ISI)** *Conditions on inservice inspection Code Cases:* Licensees may apply the ASME BPV Code Cases listed in Regulatory Guide 1.147, as incorporated by reference in paragraph (a)(3)(ii) of this section, without prior NRC approval, subject to the following:
10 CFR 50.55a(b)(5)(i)	**(ISI)** *ISI Code Case condition: Applying Code Cases.* When a licensee initially applies a listed Code Case, the licensee must apply the most recent version of that Code Case incorporated by reference in paragraph (a) of this section.
10 CFR 50.55a(b)(5)(ii)	(ISI) ISI Code Case condition: Applying different revisions of Code Cases. If a licensee has previously applied a Code Case and

TABLE 1.11-1
CODE OF FEDERAL REGULATIONS 10 CFR 50.55a REQUIREMENTS

10 CFR 50.55a Paragraphs	Conditions
	a later version of the Code Case is incorporated by reference in paragraph (a) of this section, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code Case, as authorized, or may apply the later version of the Code Case, including any NRC-specified conditions placed on its use. Licensees who choose to continue use of the Code Case during subsequent 120-month ISI program intervals will be required to implement the latest version incorporated by reference into this section as listed in Tables 1 and 2 of Regulatory Guide 1.147, as incorporated by reference in paragraph (a)(3)(ii) of this section.
10 CFR 50.55a(b)(5)(iii)	**(ISI)** *ISI Code Case condition: Applying annulled Code Cases.* Application of an annulled Code Case is prohibited unless a licensee previously applied the listed Code Case prior to it being listed as annulled in NRC Regulatory Guide 1.147. If a licensee has applied a listed Code Case that is later listed as annulled in NRC Regulatory Guide 1.147, the licensee may continue to apply the Code Case to the end of the current 120-month interval.

1.12 Code Cases

Per 10 CFR 50.55a(b)(5), Code Cases that have been determined to be suitable for use in ISI Program Plans by the USNRC are listed in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability-ASME Section XI, Division 1". The approved Code Cases in Regulatory Guide 1.147, which are being utilized by JAF, are included in Section 2.1.1. The most recent version of a given Code Case incorporated in the revision of Regulatory Guide 1.147 referenced in 10 CFR 50.55a(b)(5)(i) at the time it is applied within the ISI Program shall be used. The latest version of Regulatory Guide 1.147 incorporated into this document is Revision 18. As this guide is revised, newly approved Code Cases may be assessed for plan implementation at JAF per Paragraph IWA-2441(d) and proposed for use in revisions to the ISI Program Plan.

The use of Code Cases, other than those listed in Regulatory Guide 1.147 may be authorized by the Director, Office of Nuclear Reactor Regulation upon request pursuant to 10 CFR 50.55a(z). Code Cases not generically approved for use in Regulatory Guide 1.147, which are being utilized by JAF through associated requests for alternatives, are included in Section 8.0.

1.13 Relief Requests

In accordance with 10 CFR 50.55a, when a licensee either proposes alternatives to ASME Section XI requirements which provide an acceptable level of quality and safety, determines compliance with ASME Section XI requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, or determines that specific ASME Section XI requirements for inservice inspection are impractical, the licensee shall notify the USNRC and submit information to support the determination.

The submittal of this information will be referred to in this document as a "relief request." Relief requests for the Fifth ISI Interval and the Third CISI Interval are included in Section 8.0 of this document. The text of the relief requests contained in Section 8.0 will demonstrate one of the following: the proposed alternatives provide an acceptable level of quality and safety per 10 CFR 50.55a(z)(1)[1], compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety per 10 CFR 50.55a(z)(2), or the code requirements are considered impractical per 10 CFR 50.55a(g)(5)(iii).

Per 10 CFR 50.55a Paragraphs (z) and (g)(6)(i), the Director of the Commission will evaluate relief requests and "may grant such relief and may impose such alternative requirements as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility."

[1] 10 CFR 50.55a(a)(3) was moved to 10 CFR 50.55a(z) in the version that endorsed the 2007 Edition through the 2008 Addenda.

2.0 BASIS FOR INSERVICE INSPECTION PROGRAM

2.1 ASME Section XI Examination Requirements

As required by the 10 CFR 50.55a, this Program was developed in accordance with the requirements detailed in the 2007 Edition through the 2008 Addenda, of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWA, IWB, IWC, IWD, IWE, IWF, Mandatory Appendices, Inspection Program as referenced in IWA-2431, approved Code Cases, and approved alternatives through relief requests and Safety Evaluation Reports (SER's).

The Performance Demonstration Initiative (PDI) is an organization comprised of all US nuclear utilities that was formed to provide an efficient implementation of Appendix VIII performance demonstration requirements. The Electric Power Research Institute (EPRI) NDE Center was selected as the administrator of this program. The PDI program is administered according to the "PDI Program Description". The ISI Program implements Appendix VIII "Performance Demonstration for Ultrasonic Examination Systems," ASME Section XI 2007 Edition through the 2008 Addenda as supplemented by 10 CFR 50.55a(b)(2)(xiv) through (xvi). Appendix VIII requires qualification of the procedures, personnel, and equipment used to detect and size flaws in piping, bolting, and the reactor pressure vessel (RPV). Each organization (e.g., owner or vendor) is required to have a written program to ensure compliance with the requirements. JAF maintains the responsibility to ensure that Appendix VIII requirements are properly implemented.

For the Fifth ISI Interval, JAF's inspection program for ASME Section XI Examination Categories B-F, B-J, C-A, C-B, C-F-1, and C-F-2 will be governed by risk-informed regulations. The RI-ISI Program methodology described in Code Case N-716-1 is being used for the classification of piping welds and components under the RI-ISI Program. The RI-ISI Program scope has been implemented as an alternative to the 2007 Edition through the 2008 Addenda of the ASME Section XI Code examination program for ISI Class 1 B-F and B-J welds and ISI Class 2 C-A and C-B components, C-F-1 and C-F-2 welds. The basis for the resulting risk classification of the nonexempt ISI Class 1 and 2 piping systems at JAF is defined and maintained in the Final Report "James A. FitzPatrick Nuclear Power Plant Code Case N-716-1 Application" as referenced in Section 9.0 of this document. References to ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 have been replaced with Examination Category R-A to identify them as part of the RI-ISI Program.

The CISI Program per Subsection IWE is included in Section 6.0, "Containment ISI Plan". The CISI relief requests are included in Section 8.0 of this document.

2.1.1 ASME Section XI Code Cases

As referenced by 10 CFR 50.55a(b)(5) and allowed by USNRC Regulatory Guide 1.147, Revision 18, being the latest incorporated into this ISI Program Plan, the following Code Cases are being incorporated into the JAF ISI Program. These Code Cases have been determined by the USNRC to be acceptable alternatives to applicable parts of ASME Section XI. These Code Cases may be used by JAF without a relief request from the USNRC, provided that they are used with any identified conditions. Code Cases implemented through the relief request process are included in Section 8.0 of this document. Some of the Code Cases listed below are acceptable to the USNRC for application at JAF within the conditions imposed by the USNRC staff. Unless otherwise stated, conditions imposed by the USNRC are in addition to the requirements specified in the Code Case. Several of these Code Cases are included as contingencies, to ensure that they are available for future activities.

N-513-3 Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping, Section XI, Division 1.

Code Case N-513-3 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 18:

(1) The repair or replacement activity temporarily deferred under the provisions of this Code Case shall be performed during the next scheduled outage.

Note: JAF has an approved relief request to use Code Case N-513-4, as shown in Section 8.

N-526 Alternative Requirements for Successive Inspections of Class 1 and 2 Vessels, Section XI, Division 1

N-532-5 Repair/Replacement Activity Documentation Requirements and Inservice Inspection Summary Report Preparation and Submission Section XI, Division 1

N-586-1 Alternative Additional Examination Requirements for Class 1, 2, and 3 Piping, Components, and Supports, Section XI, Division 1

Note**:** This Code Case is implemented for Examination Categories other than R-A. N-716-1 requires that scope expansion for RI-ISI piping welds will be determined using Paragraph 6(b) of Code Case N-716-1.

N-597-2 Requirements for Analytical Evaluation of Pipe Wall Thinning, Section XI, Division 1

Code Case N-597-2 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 18:

(1) Code Case must be supplemented by the provisions of EPRI Nuclear Safety Analysis Center Report 202L-R2, April 1999, “Recommendations for an Effective Flow Accelerated Corrosion Program,” (Ref. 6), April 1999, for developing the inspection requirements, the method of predicting the rate of wall thickness loss, and the value of the predicted remaining wall thickness. As used

in NSAC-202L-R2, the term “should” is to be applied as “shall” (i.e., requirement).

(2) Components affected by flow-accelerated corrosion to which this Code Case are applied must be repaired or replaced in accordance with the construction code of record and Owner’s requirements or a later NRC approved Edition of Section III, “Rules for Construction of Nuclear Power Plant Components,” of the ASME Code (Ref. 7) prior to the value of t_p reaching the allowable minimum wall thickness, t_{min}, as specified in -3622.1(a)(1) of this Code Case. Alternatively, use of the Code Case is subject to NRC review and approval per 10CFR50.55a(z).

(3) For Class 1 piping not meeting the criteria of -3221, the use of evaluation methods and criteria is subject to NRC review and approval per 10 CFR 50.55a(z).

(4) For those components that do not require immediate repair or replacement, the rate of wall thickness loss is to be used to determine a suitable inspection frequency so that repair or replacement occurs prior to reaching allowable minimum wall thickness, t_{min}.

(5) For corrosion phenomenon, other than flow accelerated corrosion, use of the Code Case is subject to NRC review and approval. Inspection plans and wall thinning rates may be difficult to justify for certain degradation mechanisms such as MIC and pitting.

(6) The evaluation criteria in Code Case N-513-2 may be applied to Code Case N-597-2 for the temporary acceptance of wall thinning (until the next refueling outage) for moderate-energy Class 2 and 3 piping. Moderate-energy piping is defined as Class 2 and 3 piping whose maximum operating temperature does not exceed 200°F (93°C) and whose maximum operating pressure does not exceed 275 psig (1.9MPa). Code Case N-597-2 shall not be used to evaluate through-wall leakage conditions.

N-600 Transfer of Welder, Welding Operator, Brazer, and Brazing Operator Qualifications Between Owners, Section XI, Division 1

N-606-1 Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique for BWR CRD Housing/Stud Tube Repairs, Section XI, Division 1

Code Case N-606-1 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 18:

Prior to welding, an examination or verification must be performed to ensure proper preparation of the base metal, and that the surface is properly contoured so that an acceptable weld can be produced. This verification is to be required in the welding procedures.

N-613-2 Ultrasonic Examination Penetration Nozzles in Vessels, Examination Category B-D, Item No's. B3.10 and B3.90, Reactor Nozzle-to-Vessel Welds, Figures IWB-2500-7(a), (b), and (c), Section XI, Division 1

N-629 Use of Facture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials, Section XI, Division 1

N-638-6 Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique, Section XI, Division 1

Code Case N-638-6 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 18:

(1) Demonstration for ultrasonic examination of the repaired volume is required using representative samples which contain construction type flaws.

N-639 Alternative Calibration Block Material, Section XI, Division 1

Code Case N-639 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 18: Chemical ranges of the calibration block may vary from the materials specification if (1) it is within the chemical range of the component specification to be inspected, and (2) the phase and grain shape are maintained in the same ranges produced by the thermal process required by the material specification.

N-641 Alternative Pressure – Temperature Relationship and Low Temperature Overpressure Protection System Requirements, Section XI, Division 1

N-648-1 Alternative Requirements for Inner Radius Examination of Class 1 Reactor Vessel Nozzles, Section XI, Division 1

Code Case N-648-1 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 18:

In lieu of a UT examination, licensees may perform a VT-1 examination in accordance with the code of record for the Inservice Inspection Program utilizing the allowable flaw length criteria of Table IWB-3512-1 with limiting assumptions on the flaw aspect ratio.

N-666-1	Weld Overlay of Class 1, 2, and 3 Socket Welded Connections, Section XI, Division 1

Code Case N-666-1 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 18:

A surface examination (magnetic particle or liquid penetrant) must be performed after installation of the weld overlay on Class 1 and 2 piping socket welds. Fabrication defects, if detected, must be dispositioned using the surface examination acceptance criteria of the Construction Code identified in the Repair/Replacement Plan.

Note: Code Case N-666 was unconditionally approved in Rev. 17, RG 1.147.)

N-695	Qualification Requirements for Dissimilar Metal Piping Welds, Section XI, Division 1
N-702	Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1

Code Case N-702 is conditionally acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 18.

> The technical basis supporting the implementation of this Code Case is addressed by BWRVIP-108: BWR Vessel and Internals Project, “Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii,” EPRI Technical Report 1003557, October 2002 (ML-023330203) and BWRVIP-241: BWR Vessel and Internals Project, “Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii,” EPRI Technical Report 1021005, October 2010 (ML11119A041). The applicability of Code Case N-702 must be shown by demonstrating that the criteria in Section 5.0 of NRC Safety Evaluation regarding BWRVIP-108 dated December 18, 2007 (ML073600374) or Section 5.0 of NRC Safety Evaluation regarding BWRVIP-241 dated April 19, 2013 (ML13071A240) are met. The evaluation demonstrating the applicability of the Code Case shall be reviewed and approved by the NRC prior to the application of the Code Case.

Note: JAF has an approved relief request I5R-05 to utilize code case N-702, as shown in Section 8.0.

N-705	Evaluation Criteria for Temporary Acceptance of Degradation in Moderate Energy Class 2 or 3 Vessels and Tanks, Section XI, Division 1
N-716-1	Alternative Classification and Examination Requirements, Section XI, Division 1
N-730-1	Roll Expansion of Class 1 Control Rod Drive Bottom Head Penetrations in BWRs, Section XI, Division 1
N-735	Successive Inspection of Class 1 and 2 Piping Welds, Section XI, Division 1
N-747	Reactor Vessel Head-to-Flange Weld Examinations, Section XI, Division 1
N-765	Alternative to Inspection Interval Scheduling Requirements of IWA-2430, Section XI, Division 1
N-769-2	Roll Expansion of Class 1 In-Core Housing Bottom Head Penetrations in BWRs, Section XI, Division 1
N-786-1	Alternative Requirements for Sleeve Reinforcement of Class 2 and 3 Moderate Energy Carbon Steel Piping, Section XI, Division 1.
N-789	Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping for Raw Water Service, Section XI, Division 1. Code Case N-789 is conditionally acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 18. Areas containing pressure pads shall be visually observed at least once per month to monitor for evidence of leakage. If the areas containing pressure pads are not accessible for direct observation, then monitoring will be accomplished by visual assessment of surrounding areas or ground surface areas above pressure pads on buried piping, or monitoring of leakage collection systems, if available. Note: JAF has an approved relief request to use Code Case N-789-1, as shown in Section 8.0.
N-798	Alternative Pressure Testing Requirements for Class 1 Piping Between the First and Second Vent, Drain, and Test Isolation Devices, Section XI, Division 1

N-800	Alternative Pressure Testing Requirements for Class 1 Piping Between the First and Second Injection Valves, Section XI, Division 1
N-805	Alternative to Class 1 Extended Boundary End of Interval or Class 2 System Leakage Testing of the Reactor Vessel Head Flange O-Ring Leak-Detection System, Section XI, Division 1
N-823	Visual Examination, Section XI, Division 1
N-845	Qualification Requirements for Bolts and Studs, Section XI, Division 1

Additional Code Cases invoked in the future shall be in accordance with those approved for use in the latest published revision of Regulatory Guide 1.147 or 10 CFR 50.55a at that time.

2.1.2 ASME OM Code Cases

No ASME OM Code Cases are being incorporated into the JAF ISI Program Plan.

2.2 AUGMENTED EXAMINATIONS & LICENSE RENEWAL COMMITMENTS (LRC)

Augmented Examinations are not ASME Section XI requirements but are 1) additional examination areas or 2) increased inspection frequencies or a combination of both. Augmented Examinations can be requested by the Nuclear Regulatory Commission (NRC), recommended in General Electric (GE) Service Information Letters (SILs), recommended by the Boiling Water Reactor Vessel Internals Program (BWRVIP) or added by JAF management direction. Below is a summary of those examinations performed by JAF that are not specifically addressed by ASME Section XI, or the examinations that will be performed in addition to the requirements of the Code on a routine basis during the Fifth ISI Interval and the Third CISI Interval. Changes to the augmented examinations shall be in accordance with the 10 CFR 50.59 process as required.

2.2.1 "Augmented Examination of Austenitic Stainless Steel and Dissimilar Metal Welds Susceptible to Intergranular Stress Corrosion Cracking (IGSCC) (Generic Letter (GL) 88-01, NUREG-0313, Revision 2, and BWRVIP-75-A)"

Source Document: GL 88-01 "NRC Position on Intergranular Stress Corrosion Cracking in BWR Austenitic Stainless Steel Piping" Revision 2 dated January 1988 and Supplement 1 to GL 88-01 dated February 1992. EPRI Topical Report TR-1012621 (BWRVIP-75-A) "Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules" dated October 2006.

Associated Documents: NUREG-0313 Rev. 2 "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", dated January 1988. GL 84-11 "Inspections of BWR Stainless Steel Piping", dated April 1984. FSAR, Section 16.5.11 "Intergranular Stress

Corrosion Cracking (IGSCC) Program", BWRVIP-61 (EPRI TR-112076) "Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants", dated January 1999.

Background: These documents discuss the examination requirements for Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping. References to Generic Letter (GL) 88-01 within the ISI Program refer to the comprehensive commitments to all of these documents. The final SER's of BWRVIP-75 and BWRVIP-75-A revised inspection schedules were based on consideration of inspection results and service experience gained by the industry since issuance of GL 88-01 and USNRC NUREG-0313, and includes additional knowledge regarding the benefits of improved BWR water chemistry.

Since the issuance of GL 88-01, the BWR Vessels and Internals Project (BWRVIP) has been created. This BWR owners group has worked on the mitigation of IGSCC for BWR reactor vessel internal components. As part of their activities, EPRI Topical Report TR-113932, "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75) dated October 27, 1999" and EPRI Topical Report TR-1012621, "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75-A) dated October 2005" were submitted to the USNRC. Among other issues, this document proposed alternative inspection schedules for IGSCC susceptible welds. Two different inspection schedules were presented; one for plants on Normal Water Chemistry (NWC) and one for plants on effective Hydrogen Water Chemistry (HWC). The HWC schedule may be utilized if applicable performance criteria are met.

After review of BWRVIP-75 and BWRVIP-75-A, the USNRC issued SER's approving the documents with minor changes. (Letter from USNRC to Carl Terry, BWRVIP Chairman, Final Safety Evaluation of the "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)", dated May 14, 2002 and letter from USNRC to Bill Eaton, BWRVIP Chairmen, Final Safety Evaluation of the "BWR Vessel Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-76-A)", dated March 16, 2006.)

Based upon USNRC endorsement of BWRVIP-75-A, the JAF GL 88-01 (IGSCC) inspection schedule was updated to the requirements of BWRVIP-75-A except for Category A welds. (See Risked-Informed Inservice Inspection discussion below and BWRVIP discussion in Section 2.2.4).

JAF applied NobleChem (GE's patented process of applying platinum and rhodium), referred to as NMCA, in Nov. 1999 with an initial deposition of 1.18 ug/cm^2. Prior to that, JAF was operating with HWC to protect the recirculation system piping. Since Nov. 1999 JAF is considered a Category 3B NMCA plant per BWRVIP-62 Table 3-5 and is required to measure the reactor water molar ratio and catalyst loading from durability coupons. In September 2004, JAF reapplied NobleChem due to the durability coupons showing only 0.29 ug/cm^2.

The final deposition from the reapplication was 0.95 for a total remaining deposition of 1.24 ug/cm^2.

1. JAF has measured molar ratio in both the recirculation system and reactor water cleanup system sample lines and both are consistently >4:1 for operating cycles 14-23, since the initial application of NMCA.
2. The monitoring program is in compliance using a combination of deposition results on the durability monitor coupons and an artifact conservatively representative of the Recirculation system. Results of these show a reapplication was necessary during RO16. The durability monitor was placed in service in 2002 and a Recirculation ECP Flange was removed in RO15 and found to have 0.19 ug/cm^2 of noble metals remaining, which is above the recommended 0.10 ug/cm^2. Since deposition of noble metals is dependent on flow velocity and the flange is in a low area of the Recirculation system, the results of the deposition on the flange conservatively bound all of the Recirculation system. Starting in the cycle after RO16 (Nov. 2004) the durability coupon will be starting with 1.24 ug/cm^2 deposition.
3. Hydrogen availability for cycle 15 was 92% and cycle 16 was 96% as measured when reactor coolant temperature was >200° F, which is greater than the required 90% but just under the new goal of 98%. HWC is defined as available when HAS is in service, measured molar ratio > 4:1, conductivity <0.3 uS/cm and deposition as measured by the durability coupons is > 0.1 ug/cm^2.
4. For those conductivity transients > 0.3 uS/cm, JAF subtracts any hours from the HWC availability for conservatism.

In summary, JAF complied with the most restrictive requirements for "Effective NMCA" based on the initial BWRVIP-62, NRC SER and all interim positions on the open items in the sections defining "Effective NMCA" and can take credit for the table in BWRVIP-75 for NMCA inspection intervals. In order to maintain this compliance for the existing operating cycle, JAF maintained HWC availability of >90%, molar ratio >4:1 and continued to operate the durability monitor and analyze coupons for the remainder of operating cycle 18. In addition, a review and evaluation of the effectiveness of NMCA water chemistry conditions to the branch connection welds located on the Recirculation System was performed. The specific welds involved are the RHR tie-ins (24-10-130, 24-10-142, and 20-10-117) and the JPI Assemblies "A" & "B" loops. For the RHR tie-ins, the first weld off the tee connections (28"x28"x24" and 28"x28"x20" tees) is bounded by the results of the 6" Recirculation ECP flange durability results. The JPI Assembly welds were also in compliance during this time frame due to recirculation discharge flow during the initial NMCA application and continuous flow of recirculation water during operation. During the cycle leading up to RO16 refueling outage these welds were determined to comply with the requirements and criteria of BWRIVP-62 for effective NMCA water chemistry conditions and were scheduled in accordance with those criteria in the NRC's Final Safety Evaluations in BWRVIP-75-A. Note that BWRVIP-62-A was issued in November 2010. At the beginning of RO16, JAF started the applications of On-Line NobleChem (OLNC) and continued applications in

2011, 2012, 2013, 2014, 2015, 2017, and 2018. The scheduled application in 2016 was cancelled as the plant was being considered for decommissioning. Once the decision was made to continue operations, JAF obtained coupons from the durability monitor and a piece of sample line and performed an analysis to determine the noble metal disposition. The analysis showed that the noble metal deposition was greater than 0.10ug/cm^2. An OLNC campaign was completed in August of 2017. A durability monitor coupon was analyzed showing the noble metal deposition to be 0.254 ug/cm^2. This confirms that JAF continues to meet the guidance of BWRVIP-62-A and is considered a Category 3A plant.

RI-ISI guidelines have been invoked for JAF in this ISI Program Plan. Under these guidelines, ISI Class 1 and 2 piping are inspected in accordance with Code Case N-716-1 which is approved by the USNRC in Regulatory Guide (RG) 1.147. Per this code case, welds within the plant that are assigned to IGSCC Categories B through G will continue to meet existing IGSCC schedules, while IGSCC Category A welds have been incorporated into the RI-ISI Program.

Purpose: Austenitic stainless steel and dissimilar metal circumferential welds in piping four inches or larger in nominal pipe diameter which contain reactor coolant at temperature above 200°F during power operation shall be examined in accordance with the requirements of BWRVIP-75-A. Sample expansion of Categories B, C, D, or E weldments shall be in accordance with BWRVIP-75-A. Generic Letter 88-01 was issued by the NRC in 1988 to seek information regarding implementation of the new staff positions covering the industry issues with IGSCC. The staff positions were developed to cover the following subjects:

1. Materials
2. Processes
3. Water Chemistry[2]
4. Weld Overlay Reinforcement
5. Partial Replacement
6. Stress Improvement of Cracked Weldments
7. Clamping Devices
8. Crack Characterization and Repair Criteria
9. Inspection Methods and Personnel[2]
10. Inspection Schedules[2]
11. Sample Expansion[2]
12. Leak Detection
13. Reports Requirements

The NRC states in the Generic Letter “The Commission has determined that, unless appropriate remedial actions are taken, BWR plants may not be in conformance with their current design and licensing bases, including 10 CFR 50, Appendix A, General Design Criteria 4, 14, and 31.”

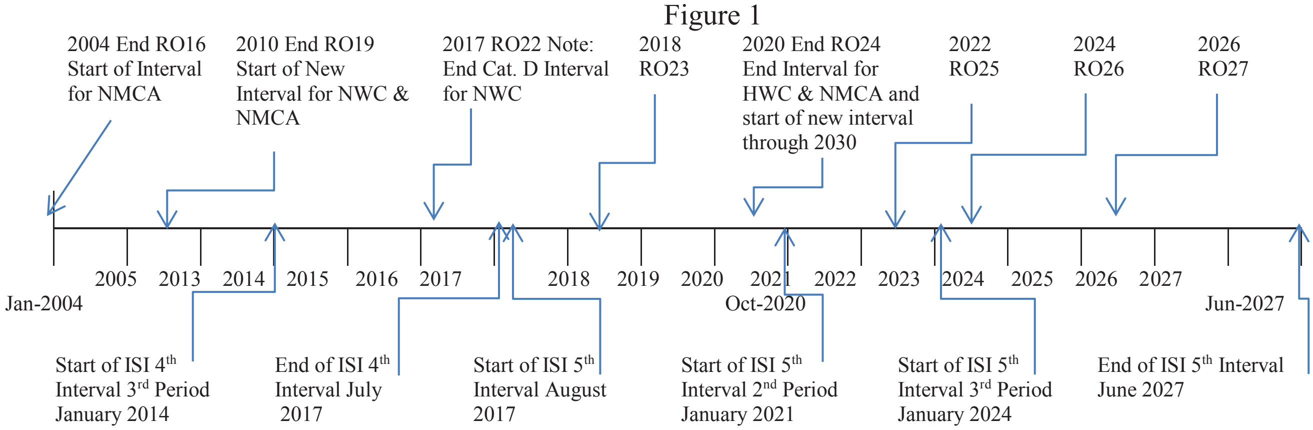
[2] With the implementation of BWRVIP-75A these commitments are superseded by the guidance in BWRVIP-75A.

This augmented examination implements the NRC Positions related to inspection of austenitic stainless steel piping in boiling water reactor environment which are susceptible to IGSCC. Because of improved water chemistry that significantly reduced the propensity for initiation and growth of IGSCC, plus improved examination procedures/techniques for the detection and sizing of IGSCC, new inspection criteria was subsequently developed and incorporated in BWRVIP-75-A. Note that the inspection frequencies of BWRVIP-75-A are used at JAF.

The following Inspection Categories were established based on the GL 88-01/NUREG-0313 requirements.

Figure 1 delineates the methodology use for the IGSCC selection process of weld examinations to be performed for Category A welds.

Figure 1



Category A – Weldments with no known cracks that are made from materials that are considered resistant to IGSCC due to their metallurgical properties. Welds joining cast pump and valve bodies to resistant materials are also considered to be Category A unless the weld material is considered susceptible. Note that all Category A welds are incorporated into the total population under Code Case N-716-1.

JAF has defined the welds in Category A by using the following suffixes:

- Category A – identifies welds, which are fabricated from resistant materials. (Total Population = 24)
- Category A* - identifies sweep-o-let welds that have been solution annealed. (Total Population – 8)

Category B - Weldments made from material that is considered susceptible to IGSCC, but the propensity for IGSCC was mitigated by stress improvement prior to two cycles of operation. JAF has no welds in this Category.

Category C - Weldments made from material that is considered susceptible to IGSCC, but the propensity for IGSCC was mitigated by stress improvement after more than two cycles of operation.

BWR Vessel and Internals Project, Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants, (BWRVIP-61) EPRI TR-112076 requires that for Category C welds, inspections shall be in accordance with NRC Generic Letter 88-01/NUREG-0313. In addition to these requirements the criteria of BWRVIP-75 and the NRC's Interim and Final Safety Evaluations of BWRVIP-75, "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules" dated September 15, 2000 and May 14, 2002 respectively, shall apply. It does not require specific review of IHSI heat treatment records but recommends that future examinations (to update IGSCC Category C welds to the current state-of-the-art) focus on the IHSI-treated welds which are "difficult to treat" such as pump-to-pipe welds, valve-to-pipe, pipe-to-tee, pipe-to-cross, ringheader end cap, and dissimilar metal welds.

JAF eligibility for reduced weld inspection frequency is based on meeting and complying with BWRVIP-61 & BWRVIP-75 conditions, if >90% HWC is not achieved for the current operating cycle a default inspection schedule using NWC conditions shall be implemented.

- Under NWC 25% every 10 years as supplemented by BWRVIP-75-A (Note 5)
- Under HWC/NMCA conditions 10% every 10 years as supplemented by BWRVIP-75-A (Note 5)

*As supplemented by Notes: 1, 2, and 3(b) (see below)
JAF has defined those welds in Category C by using the following suffixes:

- Category C-2 – identifies welds given a SI process after more than two years of operation (Total Population = 59).

- Category C* – identifies welds treated with a Resistance Heating Stress Improvement (RHSI) process after more than two years of operation (Total Population = 2).

- Category C-3 – identifies welds given an SI process after more than two years of operation and have a service stress over 1.0 S_M. Reference NUREG-0313, Rev. 2 Section 4.5 (Total Population = 3).

Category D - Weldments made from material that is considered susceptible to IGSCC where there is no mitigation by stress improvement.

JAF eligibility for reduced weld inspection frequency is based on meeting and complying with BWRVIP-61 & BWRVIP-75 conditions, if >90% HWC is not achieved for the current operating cycle a default inspection schedule using NWC conditions shall be implemented.

- NWC = 100% every 6 years

- HWC/NMCA = 100% every 10 years (at least 50% in 1st 6 years) as supplemented by Notes: 1, 2, and 3(b). Included in this category are all bimetallic nozzle weldments made with non-resistant material and 182 inconel weld butter. (Total Population = 25)

Category E - Weldments with cracks that have been overlaid with IGSCC-resistant material. Additionally, weldments with cracking that are mitigated by an effective stress improvement process may be considered Category E.

JAF eligibility for reduced weld inspection frequency is based on meeting and complying with BWRVIP-61 & BWRVIP-75 conditions, if >90% HWC is not achieved for the current operating cycle a default inspection schedule using NWC conditions shall be implemented.

- NWC = 25% every 10 years

- HWC/NMCA = 10% every 10 years as supplemented by Notes: 1, 2, and 3(b).

JAF has further defined those welds in Category E by using the following suffixes:
Category E – all welds included in this category are weld overlays (Total Population =26)

Category E* - There are no longer any welds classified in this category. Welds previously categorized as E* have been reclassified as C-2. This change is based on the examination of these components over three consecutive outages with no unacceptable indications or IGSCC type flaws found. Reference ACTS Item #98-37929.

Category F – Weldments with cracking that have not been mitigated by an effective stress improvement process. JAF has no welds in this Category.

Category G – Weldments made from material that is considered susceptible to IGSCC that have not been examined. JAF has no welds in this Category.

IGSCC Category (None) – Additionally, there is one (1) thermal sleeve crevice safe end ("A" Loop Core Spray) that is included as an augmented IGSCC inspection. Prior to the 1992 refueling outage, there were two examinations classified under this description the "A & B" side of the Core Spray System. During 1992 refueling outage, Modification F1-82-017 replaced the crevice safe end configuration on the "B" Core Spray Loop and eliminated the need to continue conducting a crevice inspection. The remaining crevice safe end ("A" Core Spray) is examined using a mock-up calibration block that simulates the configuration. During 1988 refueling outage, an inspection was performed on the "B" Core Spray crevice safe end with no evidence of defects.

<u>**Scope**</u>: The scope of this augmented program includes RCPB piping, welds and components, of four inches and larger nominal pipe size, made of stainless steel and nickel alloy. The following table provides the scope for each Category described in GL 88-01 and BWRVIP-75-A for JAF.

IGSCC Category	Total Number of Welds
A	24[1]
A*	8[1]
C-2	59
C*	2
C-3	3
D	25
E	26
N/A	1
[1]These welds are incorporated in the RI-ISI Program	

Method: Ultrasonic
Industry Code or Standards: ASME Section XI
Frequency: JAF implements the revised inspection schedules in BWRVIP-75-A.

Cat.	Weld Description	Existing Inspection Frequency of GL 88-01	Proposed Inspection Frequency (Note 1, 2, 3(b))	
			NWC	**HWC**
A	Resistant Materials	25% every 10 years at least 12% in 1st 6 years	B-F = 25% every 10 years B-J = 25% every 10 years (Note 3(a))	10% every 10 years
B	Non-Resistant Materials Stress Improved within 1st 2 years of Operation	50% every 10 years at least 25% in 1st 6 years	25% every 10 years (Notes 4 and 5)	10% every 10 years (Notes 4 and 5)
C	Non-Resistant Materials Stress Improved after 2 years of Operation	All within 2 cycles of SI, then all within 10 years at least 50% within 1st 6 years	25% every 10 years (Note 5)	10% every 10 years (Note 5)
D	Non-Resistant Materials, No Stress Improvement	Every 2 refueling cycles	100% every 6 years	100% every 10 years, at least 50% in 1st 6 years
E	Cracked – Reinforced by Weld Overlay	Every 2 refueling cycles	25% every 10 years, at least 12.5% in 1st 6 years	10% every 10 years
E	Cracked – Mitigated by Stress Improvement	Every 2 refueling cycles	100% every 6 years	100% every 10 years, at least 50% in 1st 6 years (Note 6)
F	Cracked – Inadequate or No repair	Every refueling cycle	Every Refueling Outage	Every Refueling Outage
G	Non-Resistant, Not Inspected	Next Outage	Next Outage	Next Outage

Cat.	Weld Description	Existing Inspection Frequency of GL 88-01	Proposed Inspection Frequency (Note 1, 2, 3(b))	
			NWC	HWC

1. For the examination sample percentages that are less than required by ASME Section XI for Category A welds, JAF will be implementing Code Case N-716-1 which is an approved alternative and therefore no additional alternative is required.
2. When examination sample is less than 100%, approximately 50% of the sample is required to be inspected during the first 6 years of the interval.
3. a. JAF is implementing Code Case N-716-1 and these welds are incorporated into the total population for which the element selection is used.
   b. During the selection of locations for inspection, consideration should be given regarding locations where IGSCC could be accelerated by crevice corrosion or thermal fatigue. In addition, locations having attributes that would promote IGSCC should have higher priority for inspection. The attributes that may be considered include: high carbon or low ferrite content, crevice or stagnant flow condition, evidence of weld repair, surface cold work, and high fit-up, residual and operating stresses.
4. If qualified IGSCC examinations have not been conducted, the inspection frequency for Category B welds will be 25 percent of the population every 6 years under NWC conditions, or 25 percent every 10 years under HWC conditions.
5. The licensee must ensure that an effective stress improvement was achieved. Additionally, there must have been either:
   a. a preservice (post-stress improvement) and inservice examination with a qualified procedure with no cracking identified or
   b. for welds that were stress-improved prior to publication of BWRVIP-75-A but did not receive a preservice examination, at least one examination performed with a qualified procedure after more than two operating cycles and no cracking detected.
6. If a flawed weld is stress improved and becomes Category E, a preservice examination must be performed followed by two successive inservice examinations using qualified procedures, to be performed every second refueling outage (i.e., a repeat inspection after two cycles and another inspection after two more cycles).

Sample Expansion[3]

If cracking is detected in any Category B or C weld as part of a sample examination, an additional sample of approximately the same number of welds from the category will be examined. The sample should be similar in distribution (size, system, etc.) to the original sample unless there is a technical basis for selecting a different sample. If cracking is detected in any welds in the expanded sample, all remaining welds in that Category are to be examined.

If Category D welds are examined on a sample basis and cracking is detected, the remaining Category D welds are to be examined. However, if technically justified, the sample expansion may be limited to the piping system in which cracking was initially detected.

[3] Category A is incorporated into the scope of Code Case N-716-1 and will be examined in accordance with those requirements.

All remaining Category E welds are to be examined if significant crack growth or additional cracking is detected in the initial sample. For weld overlays, significant cracking is defined as cracking that was less than 75% through-wall growing to a depth greater than 75% of wall. For cracking greater than 75% through-wall, crack growth into the effective overlay is considered significant. For SI mitigated welds, significant growth is such that a crack exceeds 10% of the circumference in length of 30% of the wall depth).

Acceptance Criteria or Standard: ASME Section XI, IWB-3640 (IWB-3514 does not apply to austenitic stainless steels and associated welds in BWR environments which are subject to stress corrosion cracking).

Regulatory Basis: GL 88-01 and the NRC Safety Evaluation for BWRVIP-75-A. The JAF Technical Requirements Manual (TRM), Section 3.4, “Reactor Coolant Systems” requires that for components within the scope of the BWR Stress Corrosion Cracking program, resistant materials will be used for new and replacement components. FSAR Section 16.10.1.5 states that resistant materials will be used for component repair/replacement activities.

Responsible Organization: Program Engineering is responsible for the development and implementation of the augmented inspection program. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.2 “Feedwater Nozzle Examinations In Accordance With U.S. NRC NUREG 0619”

Background: Boiling Water Reactor Owners’ Group (BWROG) Report GE-NE-523-A71-0594-A Revision 1, “Alternate BWR Feedwater Nozzle Inspection Requirements, May 2000,” as approved by USNRC final SER dated March 10, 2000; Boiling Water Reactor Owners’ Group (BWROG) Report GE-NE-523-A71-0594, “Alternate BWR Feedwater Nozzle Inspection Requirements, August 1999,” as conditionally approved by USNRC final SER dated June 5, 1998; and USNRC NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking”, dated November 1980.

These documents discuss the initial and current examination requirements for BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking. The alternate approach was developed and submitted to the USNRC by the BWROG. The USNRC accepted these alternate requirements in a final SER dated March 10, 2000.

JAF has removed all identified feedwater blend radii flaws, removed feedwater nozzle cladding, and installed a double piston ring, triple thermal sleeve sparger to mitigate cracking. JAF submitted a commitment change to the Feedwater Nozzle Inspection Program which was performed in accordance with NUREG-0619, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking. The periodic examination requirements for inspection of the (4) Feedwater Nozzles/Feedwater Sparger Assemblies shall be examined in accordance with the requirements of BWROG Report GE-NE-523-A71-0594, “Alternate BWR Feedwater Nozzle Inspection Requirements” in lieu of NUREG-0619, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking. This commitment change, as submitted to the NRC in NYPA Letter to NRC, JPN-99-003, dated February 18, 1999. LRA Appendix B.1.2 (Action #3 and 50 of LO-LAR-2008-0048) applies to this section and requires a review to change.

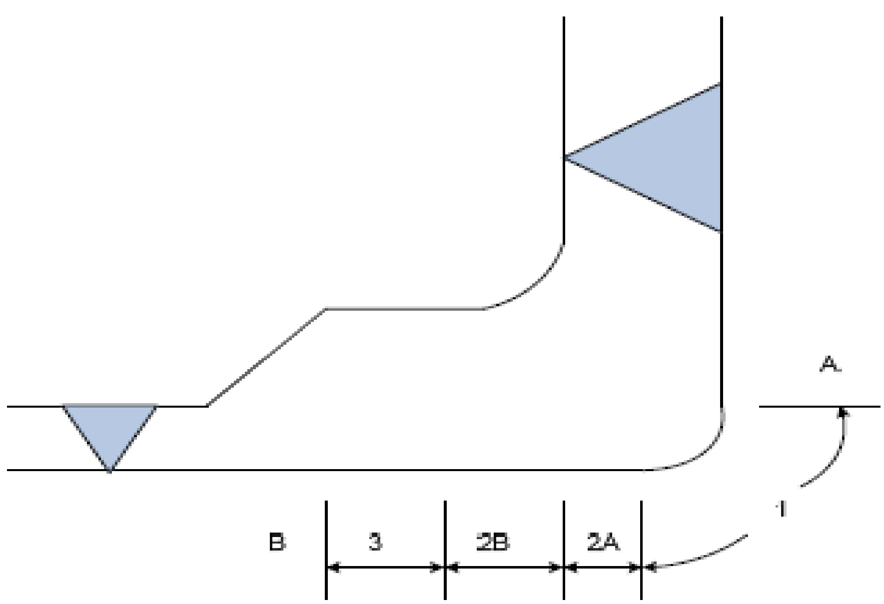
JAF performs ultrasonic examination (UT) of the four Feedwater Nozzle Inner Radii on Nozzles N4A – D (Zone 1, 2 & 3) once every 10 years in accordance with GE-NE-523-A71-0594-A, Revisions 1. JAF performs VT-3 visual examinations of the four Feedwater Spargers once every 4th refueling outage.

Source Document: The augmented examination requirements for the feedwater nozzles and spargers are contained in NUREG-0619 and BWR Owners Group (BWROG) Licensing Topical Report GE-NE-523-A71-0594, Revision 1, August 1999, Table 6-1.

Associated Documents: General Electric document GE-NE-523-A71-0594-A, Revision 1, May 2000, "Alternate BWR Feedwater Nozzle Inspection Requirements," and the NRC Final Safety Evaluation of BWR Owner's Group Alternate Boiling Water Reactor (BWR) Feedwater Nozzle Inspection. BWROG-TP-14-012 (July 2014), Feedwater Nozzle Inspection Frequency 2014, Interference Fit Spargers-Definition Clarification

Purpose: NUREG-0619 was issued by the NRC in November 1980 and described a cracking phenomenon of BWR RPV Feedwater (FW) nozzle and CRD nozzle inside radius sections. As a result of enhanced technology and more sophisticated techniques for stress and fracture mechanics analysis, the examination of the FW Nozzle Blend Radius is now performed in accordance with NRC approved guidance of GE-NE-523-A71-0594-A Rev.1.

Scope: The scope of this augmented examination program section includes UT of all four of the FW nozzle bores and inside radius sections as depicted in Figure 1 and VT-3 visual examinations for the FW spargers.



The volumetric UT examination region begins at the inside radius-to-vessel intersection point (A). The examination region ends at the point on the inner diameter (ID) corresponding to the point on the outer diameter (OD) where the taper on the nozzle thickness starts at (B).

Figure 1

Method: Volumetric UT examination will be performed on the FW nozzle inside radius sections and VT-3 visual examinations will be performed on the FW spargers.

Industry Code or Standards: ASME Section XI

Frequency: FW Nozzle Zones 1, 2, and 3 are examined once every 10 years. The Feedwater Spargers are internal to the reactor pressure vessel and shall be visually examined every fourth refueling outage.

Acceptance Criteria or Standard: ASME Section XI, IWB-3000 and the supplemental guidance provided in GE-NE-523-A71-0594-A, Rev. 1.

Regulatory Basis: The JAF Final Safety Analysis Report (FSAR), Section 16, 16.10.1.3 “BWR Feedwater Nozzle”, provides the basis for inspection of the feedwater nozzles and has statements, which support the use of these examinations that will be performed in accordance with ASME Section XI, Appendix VIII to achieve the level of confidence needed as specified in NUREG-0619.

Responsible Organization: Program Engineering is responsible for the development and implementation of the augmented inspection program. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.3 **“Core Spray Augmented Examinations”**

Background: Core Spray Pump discharge piping experienced vibration and in 1991 a calculation was completed to determine the acceptability of the piping vibration. The vibration occurs during the routine pump testing operation. The concern was whether the vibration could result in piping failure. Vibration amplitudes at seven (7) location on the discharge piping were measured on each loop. The conclusion was that the vibration would not cause pipe failure. However, a selection of welds was identified for examination to confirm the results.

Source Document: JAF-CALC-CSP-00327

Associated Documents: N/A

Purpose: Monitoring core spray pump discharge piping vibration

Scope: The following welds were selected for examination in order to monitor piping vibration:

Core Spray Loop A	**Core Spray Loop B**
12-14-724	12-14-823
12-14-734	12-14-834

12-14-750	12-14-851
8-14-779	8-14-878
8-14-780	8-14-879
	10-14-884A

Method: Volumetric and Surface

Industry Code or Standards: ASME Section XI

Frequency: Based on previous examinations and the results of the calculation, no additional examinations are required. This is kept for historical purposes.

Acceptance Criteria or Standard: IWC-3514

Regulatory Basis: N/A
Responsible Organization: Program Engineering is responsible for the development and implementation of the augmented inspection program. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.4 **"Augmented Main Steam and Feedwater Inspection Program (HELB)"**

Background: Technical Requirements Manual (TRM) 3.4.A states, in part, "Several locations on the main steam lines and feedwater lines are not restrained to prevent pipe whip in the event of pipe failure at these locations. The physical layout within the drywell precludes restraints at these points. Unrestrained high stress areas have been identified in these lines where breaks could result in pipe whip such that the pipe could impact the primary containment. Augmented inservice inspection shall be performed during each inspection period.

Source Document: Technical Requirements Manual Section 3.4.A

Associated Documents: JAF-RPT-03-00289 "JAFNPP Augmented Main Steam and Feedwater High Stressed Weld Inspection Program", FSAR Section 16.3.2.2 "Excessive Pipe Movement", and EPRI TR-1006937 "Extension of the EPRI Risk Informed ISI Methodology to Break Exclusion Region Programs," dated April 4, 2002.

Purpose: Based on the above requirements, an augmented inspection program is implemented on the Main Steam and Feedwater systems. Currently there are a total of 22 welds on the Feedwater lines and 12 welds on the Main Steam Lines within the drywell.

Scope: The following table identifies the system and number of welds selected.

Main Steam		Feedwater	
MSK-3031	MSK-3032	MSK-3033	MSK-3034
24-29-541	24-29-588	12-34-368	18-34-420
24-29-557	24-29-584	12-34-369	12-34-403
24-29-552	**24-29-589**	12-34-370	12-34-404
24-29-553	24-29-605	12-34-371	12-34-405

Main Steam		Feedwater	
MSK-3031	**MSK-3032**	**MSK-3033**	**MSK-3034**
24-29-559	24-29-636	**12-34-375**	**12-34-409**
24-29-572	**24-29-583**	12-34-373	12-34-406
		12-34-372	12-34-407
		18-34-374	18-34-408
		18-34-382	18-34-416
		18-34-389	18-34-423
		18-34-391	18-34-425

Based on the results of 50.59 Evaluation JAF-SE-03-004 the main steam sample inspection population will consist of 2 welds. The 2 welds selected shall be among the high stressed welds evaluated as medium risk significant ranking (Table 1 from JAF-RPT-03-00289) and welds within the highest stress group based on MS & FW Crack Detection/Stress Levels. The inspection for cause method shall be the method evaluated for the specific type of damage mechanism. The feedwater sample inspection population will consist of 4 welds. The four welds selected shall be among the high stressed welds evaluated as high and medium risk significant ranking (Table 1 from JAF-RPT-03-00289), also welds with multiple damage mechanisms and within the highest stress group based on MS & FW Crack Detection/Stress Levels. The inspection for cause method shall be the method evaluated for the specific type of damage mechanism. The welds selected are those identified in **bold** text above.

Method: Volumetric

Industry Code or Standards: ASME Section XI

Frequency: Each inspection interval

Acceptance Criteria or Standard: ASME Section XI

Regulatory Basis: TRM 3.4.A and EPRI TR-1006937

Responsible Organization: Program Engineering is responsible for the development and implementation of the augmented inspection program. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.5 **"Containment Inspection Program (Torus Exterior Shell)"**

Background: Ultrasonic thickness measurements shall be performed from the exterior surface of the Torus Shell in accordance with JAF calculation JAF-CALC-16-00008 Rev. 0 "FitzPatrick Torus Corrosion Allowance." These examinations are being performed in support of the Torus Preservation Program and are not required based on the IWE Containment Inspection Program.

Examinations previously performed from the Torus exterior shell at designated HPCI and RCIC locations have been eliminated based on Root Cause Evaluation RC-CR-2005-2593 Rev. 2 CA-18 & 19 along with CR-JAF-2007-2149 CA-1.

Source Document: JAF-CALC-16-00008 Rev. 0 “FitzPatrick Torus Corrosion Allowance”

Associated Documents: RC-CR-2005-2593 Rev. 2 “Inspectors discovered a TORUS leak in the vicinity of a TORUS Support between Bays A and P”, CR-JAF-2007-2149 “Investigate and Correct the condition regarding ultrasonic examination not being performed of the Torus exterior near the HPCI and RCIC.”

Purpose: Verify structural integrity of the Torus

Scope: Exterior of the Torus at the following specific areas
B-3-2, B-4-1, H-1-2, H-4-1, K-3-1, K-3-2, O-2-1, O-3-2, and O-3-3

Method: Ultrasonic Thickness Measurements

Industry Code or Standards: N/A

Frequency: Every two refueling outages

Acceptance Criteria or Standard: JAF-CALC-05-00037

Regulatory Basis: Technical Specifications 4.6.2.1(e)

Responsible Organization: Program Engineering is responsible for the development and implementation of the augmented inspection program. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.6 Main Steam System

Background: Augmented examinations of the Main Steam System supports (Augmented Category 2A) has been removed from the fifth Inservice Inspection Interval. Reference IR-04037760.

2.2.7 (LRC) “BWR Feedwater Nozzle”

Background: This is an existing program at JAF as described in FSAR 16.10.1.3 “BWR Feedwater Nozzle.” Under this program, JAF has removed all identified feedwater blend radii flaws, removed feedwater nozzle cladding, and installed a double piston ring, triple thermal sleeve sparger to mitigate cracking. This program implements enhanced inservice inspection of the feedwater nozzles in accordance with the requirements of ASME Section XI, Subsection IWB and the recommendation of General Electric (GE) NE-523-A71-0594 to monitor the effects of cracking on the intended function of the feedwater nozzles.

Source Document: LRA Appendix B, B.1.3 “BWR Feedwater Nozzle” (P-18369 LO-LAR-2008-0048 CA-32)

Associated Documents: GE-NE-523-A71-0594 "Alternate BWR Feedwater Nozzle Inspection Requirements", JAF-RPT-09-LR003, dated April 18, 2011

Purpose: See 2.2.2

Scope: See 2.2.2

Method: See 2.2.2

Industry Code or Standards: See 2.2.2

Frequency: See 2.2.2

Acceptance Criteria or Standard: See 2.2.2

Regulatory Basis: FSAR 16.10.1.3 "BWR Feedwater Nozzle" (P-18378 LO-LAR-2008-0048 CA-41)

Responsible Organization: See 2.2.2

2.2.8 **(LRC) "Containment Inservice Inspection"**

Background: This is an existing program that is being credited to the aging management during the period of extended operation. This is a plant-specific program encompassing requirements for the inspection of Class MC pressure-retaining components (Primary Containment) and their integral attachments in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE. This program manages loss of material for the primary containment and its integral attachments. The Containment Inservice Inspection Program, which is supplemented by the Containment Leak Rate Program is credited with the following aging effects:

- Loss of material and cracking for carbon steel components (AMC-01)
- Loss of material and cracking for carbon steel containment penetration components (AMC-01)
- Cracking for stainless steel components (AMC-01).

The primary inspection method for the primary containment and its integral attachments is by visual examination either directly or remotely. Visual examinations are performed either directly or remotely with sufficient illumination and resolution suitable for local environment to assess the general conditions that may affect structural integrity or leak tightness. Results are compared, as appropriate to baseline data and other previous inspection results, and acceptance criteria of ASME Section XI, Subsection IWE for evaluation of degradation. The Containment Inservice Inspection program is consistent with GALL Section XI.S1, ASME Section XI, Subsection IWE.

The primary containment is a General Electric Mark 1 pressure suppression containment system. The system consists of a drywell (housing the reactor vessel and reactor coolant recirculation loops), a pressure suppression chamber (housing a water pool), and the connecting vent system between the drywell and the water pool, isolation valves, and containment cooling systems. The code of construction for the containment structure is the ASME Section III, 1968 Edition including the 1968 Summer Addenda.

Source Document: LRA, Appendix B, B.1.16.1 "Containment Inservice Inspection"

Associated Documents: ASME Section XI, 2007 Edition through the 2008 Addenda, JAF-RPT-09-LR161, dated April 11, 2011, NUREG-1905, Safety Evaluation Report Related to the License Renewal of James A. FitzPatrick Nuclear Power Plant, JAF-CALC-PC-04436, EC 62569.

Purpose: See 6.0

Scope: See 6.0

Method: See 6.0

Industry Code or Standards: ASME Section XI, Subsection IWE

Frequency: See 6.0

Acceptance Criteria or Standard: See 6.0. ASME Section XI, IWE-3510 and IWE-3530, as well as any applicable NDE Procedures (ER-AA-335 Series) for acceptance criteria. Reference JAF-CALC-PC-04436 for thickness readings.

Regulatory Basis: FSAR, 16.10.1.17 "Containment Inservice Inspection"

Responsible Organization: Program Engineering is responsible for the development and implementation of the AMP. Responsible Individual is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.9 **(LRC) "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"**

Background: This is an existing program that is being credited to the aging management during the period of extended operations. This program consists of periodic volumetric and visual examinations of components for assessment, identification of signs of degradation, and establishment of corrective actions.

Source Document: LRA, Appendix B, B.1.16.2 "Inservice Inspection Program" (P-18279 LO-LAR-2008-0048 CA-56)

Associated Documents: ASME Section XI, 2007 Edition through the 2008 Addenda, JAF-RPT-05-LRD02

Purpose: This program includes Volumetric and Visual VT-1, VT-2, and VT-3 examinations performed to manage cracking, loss of fracture toughness and loss of material in Class 1, 2, and 3 piping and components exposed to reactor coolant, steam and treated water environments.

Scope: See Section 2.1

Method: See Section 2.1

Industry Code or Standards: ASME Section XI

Frequency: See Section 2.1

Acceptance Criteria or Standard:

Regulatory Basis: FSAR, 16.10.1.18 "Inservice Inspection Program"

Responsible Organization: Program Engineering is responsible for the development and implementation of the AMP. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.10 **(LRC) "BWR Stress Corrosion Cracking"**

Background: This is an existing program that is being credited to the aging management during the period of extended operations. This program manages intergranular stress corrosion cracking (IGSCC) in reactor coolant pressure boundary piping and piping components made of stainless steel (SS) and nickel based alloy components as delineated in NUREG-0313, Rev. 2 and Generic Letter (GL) 88-01 and its Supplement 1. These welds are volumetrically examined per 2.2.1 of this section. This program will be enhanced to clarify that components within the scope of this program, will have resistant materials used for new and replacement components. This includes low carbon stainless piping and stainless steel weld material limited to a maximum carbon content 0.035wt.% and a minimum ferrite content of 7.5%. JAF has taken actions to prevent IGSCC and will continue to use material resistant to IGSCC for component replacements and repairs following the recommendations delineated in NUREG-0313, Generic Letter 88-01, and the staff-approved BWRVIP-75-A report.

Source Document: LRA, Appendix B, B.1.5 "BWR Stress Corrosion Cracking" (P-18371 LO-LAR-2008-0048 CA-34)

Associated Documents: JAF-RPT-09-LR005, dated April 12, 2011

Purpose: See 2.2.1

Scope: See 2.2.1

Method: See 2.2.1

Industry Code or Standards: See 2.2.1

Frequency: See 2.2.1

Acceptance Criteria or Standard: See 2.2.1

Regulatory Basis: FSAR, 16.10.1.5 "BWR Stress Corrosion Cracking"

Responsible Organization: See 2.2.1

2.2.11 **(LRC) "BWR Control Rod Drive Return Line Nozzle"**

Background: This is an existing program that is being credited to the aging management during the period of extended operations. This program provides monitoring of the N9 nozzle for cracking through station ISI procedures based on ASME Section XI requirements. JAF has capped the N9 nozzle to mitigate fatigue cracking. In 2000, a structural weld overlay was installed over a crack in the CRD return line nozzle-to-cap weld. The nickel-based Alloy 52 weld metal used in the overlay is highly resistant to stress corrosion cracking, which was determined to be the cause of the cracking. The program performs volumetric examination of the nozzle-to-vessel weld and the nozzle inner radius. The N9 nozzle-to-cap weld is also volumetrically examined.

Source Document: LRA, Appendix B, B.1.2 "CRD Return Line Nozzle" (P-18343 LO-LAR-2008-0048 CA-3, 50)

Associated Documents: JAFP-06-0109, dated July 31, 2006, Generic Letter 88-01 "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping."

Purpose: Perform ultrasonic and visual examinations to verify the structural integrity of the CRD Return Line nozzle and cap.

Scope: CRD Return line nozzle to cap weld overlay, CRD Return line nozzle-to-vessel welds, and the CRD Return line nozzle blend radius and adjacent vessel wall.

Method: Ultrasonic and Enhanced Visual Examination (1/2 mil wire resolution)

Industry Code or Standards: ASME Section XI

Frequency: Every Inspection Interval

Acceptance Criteria or Standard: ASME Section XI

Regulatory Basis: FSAR, 16.10.1.2 "CRD Return Line Nozzle"

Responsible Organization: Program Engineering is responsible for the development and implementation of the AMP. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.12 **(LRC) "BWR Penetrations Program"**

Background: This is an existing program that is being credited to the aging management during the period of extended operations. This program manages the effects of cracking of reactor vessel instrumentation penetrations (nozzles) exposed to reactor coolant through water chemistry and inservice inspections. This program incorporates the inspection and evaluation recommendations of BWRVIP-27-A "BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines" and "BWRVIP-49-A "Instrument Penetration Inspection and Flaw Evaluation Guidelines" and well as the water chemistry recommendations of BWRVIP-130, "BWR Vessel and Intervals Project BWR Water Chemistry Guidelines."

Source Document: LRA, Appendix B, B.1.4 "BWR Penetrations" (P-18370 LO-LAR-2008-0048 CA-33)

Associated Documents: BWRVIP-27-A "BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines", "BWRVIP-49-A, "Instrument Penetration Inspection and Flaw Evaluation Guidelines", JAF-RPT-09-LR023, dated May 18, 2011 and JAF-RPT-05-LRD02.

Purpose: Manage the effects of cracking of reactor vessel instrumentation nozzles

Scope: Beltline instrumentation nozzles and other instrumentation nozzles.

Method: Visual VT-2 in accordance with ASME Section XI, Table IWB-2500-1, Category B-P. JAF performs a surface examination, on the safe-end and safe-end extension welds, every 10 years until such time that a volumetric inspection technique is developed.

Industry Code or Standards: ASME Section XI, 2007 Edition through the 2008 Addenda

Frequency: Every Refueling Outage

Acceptance Criteria or Standard: IWB-3522

Regulatory Basis: FSAR, 16.10.1.4 "BWR Penetrations"

Responsible Organization: Program Engineering is responsible for the development and implementation of the AMP. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.13 **(LRC) "Reactor Head Closure Studs"**

Background: This is an existing program that is being credited to the aging management during the period of extended operations. This program is implemented through station procedures based on the examination and inspection requirements specified in ASME Section XI, Table IWB-2500-1 and preventive measures described in NRC Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessel Closure Studs."

Source Document: LRA, Appendix B, B.1.23 "Reactor Head Closure Studs" (P-18370 LO-LAR-2008-0048 CA-43)

Associated Documents: ASME Section XI, 2007 Edition through the 2008 Addenda, JAF-RPT-09-LR-023, dated May 18, 2011

Purpose: This program includes volumetric examinations performed to manage cracking, loss of fracture toughness and loss of material in reactor head closure studs and visual examination of the nuts and washers.

Scope: Reactor Pressure Vessel Closure Head Studs, Nuts and Washers.

Method: Volumetric

Industry Code or Standards: ASME Section XI

Frequency: Once per 10-year interval per ASME Section XI, Table IWB-2500-1, Category B-G-1.

Acceptance Criteria or Standard: ASME Section XI, IWB-3515 or IWB-3517

Regulatory Basis: 10 CFR 50.55a "Codes and Standards", FSAR 16.10.1.25 "Reactor Head Closure Studs"

Responsible Organization: Program Engineering is responsible for the development and implementation of the AMP. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.14 (LRC) "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"

Background: The purpose of the Thermal Aging and Neutron Irradiation Embrittlement of CASS Program is to assure that reduction of fracture toughness due to thermal aging and reduction of fracture toughness due to radiation embrittlement will not result in loss of intended function during the period of extended operation. This program evaluates CASS component in the reactor vessel internals and requires non-destructive examinations as appropriate. Currently there are no required augmented inspections, however JAF will continue to monitor BWRVIP-234-A and implement any mandatory actions resulting from that review as part of the CASS program.

Source Document: LRA, Appendix B, B.1.28 "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)" (A-18359 LO-LAR-2008-0048 CA-18)

Associated Documents: JAF-RPT-09-LR026, dated December 6, 2010, ER-JF-331-1001 "JAF Reactor Vessel & Internals Program Bases Document", BWRVIP-234-A "Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Interval" dated December 2009

Purpose: To determine whether the high temperature and neutron fluence will affect the CASS components within the reactor vessel interior.

Scope: The following CASS material components susceptible to thermal aging and neutron irradiation embrittlement and subject to loss of fracture toughness include:

- Orificed Fuel Supports
- CRD Guide Tube Bases
- Core Spray Sparger Elbows
- Jet Pump Assemblies; transition piece, inlet, throat, and diffuser collar

Method: Evaluation and supplemental inspection using enhanced visual VT-1 examination method

Industry Code or Standards: N/A

Frequency: Based on BWRVIP-234-A, no augmented inspections of BWR CASS components are required based on the following:

- All BWR CASS components have ferrite levels below the level for which aging embrittlement is a concern
- CASS Jet Pump assembly components and Orificed Fuel Supports meet the NRC approved fracture toughness threshold value
- End of Life (EOF) fluence level at the Control Rod Guide (CRD) Tube base and Core Spray Sparger nozzle elbows is less than the threshold value for toughness loss.

Acceptance Criteria or Standard: Detected flaws are evaluated in accordance with IWB-3500. Flaw tolerance evaluation for components with ferrite content up to 25% are performed according to the principles associated with IWB-3640 procedures for submerged arc welds regarding the Code restriction of 20% ferrite in IWB-3641(b)(1).

Regulatory Basis: FSAR, 16.10.1.31 "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"

Responsible Organization: Program Engineering is responsible for the development and implementation of the AMP. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.15 (LRC) "BWR Vessel ID Attachment Welds"

Background: This is an existing program that is being credited to the aging management during the period of extended operations. This program incorporates the recommendations of BWRVIP-48-A and monitoring and control of reactor coolant water chemistry in accordance with guidelines of BWRVIP-190 to ensure the long-term integrity and safe operation of reactor vessel inside diameter (ID) attachment welds and support pads. This is implemented through station procedures that are part of inservice inspection and incorporates the requirements of ASME Section XI.

Source Document: LRA, Appendix B, B.1.6 "BWR Vessel ID Attachment Welds" (P-18372 LO-LAR-2008-0048 CA-35)

Associated Documents: ASME Section XI, 2007 Edition through the 2008 Addenda, BWRVIP-48-A "BWR Vessel and Internals Project Vessel ID Attachment Weld

Inspection and Flaw Evaluation Guidelines", JAF-RPT-09-LR006, dated April 20, 2010, and ER-JF-331-1001 "JAF Reactor Pressure Vessel & Internals Program Bases Document"

Purpose: This program includes visual EVT-1, VT-1 and VT-3 examinations performed to manage cracking, loss of fracture toughness and loss of material in reactor vessel internal attachment welds. This program is implemented by the Inservice Inspection (ISI) and the Reactor Vessel Internals Programs.

Scope: Jet Pump Riser Braces, Core Spray Piping, Steam Dryer Support, Feedwater, Steam Dryer Support and Hold Down Brackets, Guide Rod, Feedwater Sparger, and Surveillance Sample Holder Brackets.

Method: Visual Examination EVT-1, VT-1 and VT-3

Industry Code or Standards: ASME Section XI

Frequency: BWRVIP-48-A, Table 3-2.

Acceptance Criteria or Standard: ASME Section XI, IWB-3520

Regulatory Basis: 10 CFR 50.55a "Codes and Standards", FSAR 16.10.1.6 "BWR Vessel ID Attachment Welds"

Responsible Organization: Program Engineering is responsible for the development and implementation of the AMP. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.16 **(LRC) "Lubrite Sliding Supports"**

Background: As part of the Inservice Inspection Program the performance of inspections of the lubrite surfaces on the torus support saddles is completed. These inspections are performed to confirm the absence of aging effects for the lubrite surfaces.

Source Document: LRA, Appendix B, B.1.16.2 "Inservice Inspection" (P-18379 LO-LAR-2008-0048 CA-42)

Associated Documents: LRA, Appendix B, B.1.27.2 "Structures Monitoring"

Purpose: Since there were no aging effects requiring management identified for lubrite sliding supports, JAF committed to enhance the ISI Program to inspect the torus support saddles to confirm the absence of aging affects for the period of extended operation.

Scope: Torus Support Saddles

Method: Visual Examination, VT-3

Industry Code or Standards: ASME, Section XI

Frequency: Every Inspection Interval

Acceptance Criteria or Standard: Section XI, IWF-3410

Regulatory Basis: FSAR, 16.10.1.30 "Structures Monitoring – Structures Monitoring Program"

Responsible Organization: Program Engineering is responsible for the development and implementation of the AMP. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.17 **"Containment Inspection Program (Concrete to steel shell interface)"**

Background: JAF does not have a moisture barrier in the drywell around the concrete to steel shell interface. ASME Section XI IWE-1241 requires areas that are subject to accelerated degradation and aging require augmented examinations identified in Table IWE-2500-1, Examination Category E-C. IWE-1241(a) lists concrete-to-steel shell or liner interfaces as an area subject to accelerated degradation and aging.

Source Document: OE-NOE-2016-145.

Associated Documents: RIS-16-08.

Purpose: Monitor areas subject to accelerated degradation and aging.

Scope: Table IWE-2500-1, Examination Category E-C, Item Nos. E4.11 and E4.12.

Method: VT-1 and Ultrasonic Thickness measurements.

Industry Code or Standards: ASME Section XI, IWE-1241.

Frequency: 100% each inspection period until the areas examined remain essentially unchanged for the next inspection period. (IWE-2500-1, Examination Category E-C, Footnote 1).

Acceptance Criteria or Standard: IWE-3520.

Regulatory Basis: 10 CFR 50.55a "Codes and Standards"

Responsible Organization: Program Engineering is responsible for the development and implementation of the augmented inspection program. Design Engineering is responsible for evaluating conditions of degradation for acceptance or corrective action.

2.2.18 **"Early Visual Leakage Examination for Instrumentation Nozzles"**

Background: As required by NER NC-18-005-Y, schedule early visual inspections each refueling outage for all BWR RPV instrument nozzles until repair and non-visual NDE techniques are developed (Action 4 NER NC-18-005-Y and Limerick RCE action 4007992-43).

Source Document: NER NC-18-005-Y.

Associated Documents: None.

Purpose: Performing early visual leakage examinations, prior to the system leakage required prior to start up, will allow early identification of indications requiring repair. Although leakage is not a nuclear safety concern, this is a risk management issue that could extend the duration of outages.

Scope: Bare Metal Visual Examinations on BWR RPV Instrument Nozzles

Method: Visual Examination

Industry Code or Standards: N/A

Frequency: Every Outage

Acceptance Criteria or Standard: No signs of leakage

Regulatory Basis: None

Responsible Organization: Program Engineering is responsible for scheduling and implementation of the visual examinations.

2.3 System Classification and Flow Diagrams

The Inservice Inspection classification process described below resulted in system classifications indicating ASME Class 1, 2, 3, and MC components. Class boundaries were developed and are shown on the ISI Flow Diagrams (ISI-FM & FB series drawings) listed on Table 2.3-1 and available in the Electronic Data Management System (EDMS).

For ASME Class 1 components, the requirements of Subsection IWB apply; for ASME Class 2 components, the requirements of Subsection IWC apply; for ASME Class 3 components, the rules of Subsection IWD apply; and for ASME Class MC components, the requirements of Subsection IWE apply. The rules of Subsection IWF for component supports apply to ASME Classes 1, 2, 3, and MC.

The ISI-FM & FB(s) have ISI Class designations on piping lines showing ISI Class and ISI Class Breaks delineating the ISI Boundary.

Class 1, 2, and 3 boundary breaks at valves wholly within a particular plant system, or class breaks between two different plant systems, are taken so the classification of the valve is that of the higher of the two classifications. The first weld after the class break on the side of the valve with the lower classification is designated at the lower classification.

2.3.1 Inservice Inspection Boundary Development

In accordance with ASME Section XI, Paragraph IWA-1400, it is the Owner's responsibility to determine the appropriate code classes for each component of the power plant and to identify the system boundaries for each class of components. Section XI refers to 10CFR50 for the classification criteria. 10CFR50.2 defines the reactor coolant pressure boundary (RCPB) and 10CFR50.55a requires that it be classified as Class 1. 10CFR50.55a(d)(1) and (e)(1) Footnote 7 states that guidance for classifications may be found in Regulatory Guide 1.26 and Section 3.2.2 of NUREG-0800.

JAF was designed and constructed prior to the issuance of the Regulatory Guides, therefore, systems and components were not classified under the quality group classifications defined in Reg. Guide 1.26. Also, JAF does not fall under or required to meet the Standard Review Plan requirements.

The Code Classifications for JAF ISI-FM/FB Series drawings were prepared using the guidance of Reg. Guide 1.26 and NUREG-0800 and includes only those systems which are important to safety that contain water, steam, or radioactive materials. JAF is not committed to this standard applying Reg. Guide 1.26 for ISI Classifications which is consistent with other utilities' programs. There are some discrepancies between this Reg. Guide and JAF classifications. For instance, the Main Steam and Feedwater piping beyond the containment isolation valves and reactor building wall is not safety-related per the Q-List of FSAR and is therefore classified as Quality Group D. The Reg. Guide and SRP require that the piping up to the turbine stop valve and up to the first valve capable of automatic closure in the feedwater lines be classified as Quality Group B.

ISI Class boundaries designating Class 1, 2, 3, and MC piping and components are controlled and depicted on the ISI-FM/FB Series drawings listed in Table 2.3-1. Welds, components and component supports that are subject to inservice inspection are shown on the JAF Piping Isometric Drawings listed in Table 2.4-1. Pursuant to 10 CFR 50.55a, the inservice inspection requirements of ASME Section XI have been assigned to these components within the constraints of existing plant design.

2.3.2 Component and System Quality Group Classifications and Boundary Classifications

Sections 10 CFR 50.55a(c), (d), and (e) reference either 10CFR50.2 or Regulatory Guide 1.26 for the classification of Quality Group A, Quality Group B and C components, respectively. Footnote 7 of 10CFR50.55a specifically invokes Regulatory Guide 1.26. 10CFR50.2 provides criteria for the classification of Quality Group A components.

In accordance with 10CFR50.55a(c), the Class 1 boundaries are described in 10CFR50.2 which defines the reactor coolant pressure boundary as follows:

"All those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

(1) Part of the reactor coolant system, or
(2) Connected to the reactor coolant system, up to and including any and all of the following:
    (i) The outermost containment isolation valve in system piping which penetrates primary containment,
    (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,
    (iii) The reactor coolant system safety and relief valves.

For nuclear power reactors of direct boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in main steam and feedwater piping."

(1) Quality Group A (ASME Code Class 1)
    Quality Group A system boundaries were developed based on 10CFR50.2 and the JAF FSAR and apply to the reactor coolant pressure boundary components. The Reactor Coolant system includes a single cycle, forced circulation, General Electric Boiling Water Reactor.

(2) Quality Group B (ASME Code Class 2)
    Quality Group B system boundaries were developed based on Regulatory Guide 1.26 and the JAF FSAR and apply to those components of the Reactor Coolant System not classified as Quality Group A, (ASME Code Class 1), and that are safety related.

(3) Quality Group C (ASME Code Class 3)
    Quality Group C system boundaries were developed based on Regulatory Guide 1.26 and the JAF FSAR and apply to those components that are not classified as Quality Group A or B, (ASME Code Class 1 or 2), and that are safety-related.

(4) Quality Group D (Non-Nuclear Safety-Related)
    Quality Group D is non-ASME Code Class and are those components not related to nuclear safety.

2.3.3 Regulatory Guide 1.26

10CFR50.55a does not specifically define the boundaries for Class 2 and 3 piping systems, but in Footnote 7 it refers to Regulatory Guide 1.26 and NUREG-0800, Section 3.2.2 for guidance. For JAF, guidance is taken from Regulatory Guide 1.26, which provides general boundary definitions for systems containing water, steam or radioactive waste. It should be noted that Regulatory Guide 1.26 specifically states that it does not address other systems such as "…instrument and service air, diesel engine and its generators and auxiliary support systems, diesel fuel, emergency and normal ventilation, fuel handling, and radioactive waste management systems".

2.3.4 Exceptions and Exclusions to the ISI Program Plan

2.3.4.1 Systems/Components and/or Piping Exclusions

Standby Liquid Control (SLC): Based on the operational parameters of the Class 1 portion of the Standby Liquid Control System (SLC), which is not exposed to boron, the requirements of IWA-5241(f) shall not be applied.

Class 2 Control Rod Drive Systems: All pressure tests shall be performed during reactor startup and/or shutdown during a refueling and/or maintenance outage. Reference CRD testing requirements, dated, February 9, 1999, "ISI Pressure Testing of the CRD System" and JAF-ICD-CRD-04093, Rev. 0.

Main Steam System: JAF has optionally upgraded portions of the Main Steam system to an augmented Class 2 category (2A) for the purposes of including the supports in the inspection population. This includes the first two supports beyond the MSIV(s). Main Steam Class 2 designated welds listed within the program are for informational purposes only and are not included in the selection, prorated counts or percentages for program compliance.

Emergency Diesel Generator (EDG): JAF has optionally upgraded portions of the EDG system to an augmented class category (3A) for the purposes of pressure testing only.

2.3.5 ASME Section XI

In accordance with 10CFR50.55a, the inservice inspection of Class 1, 2, 3, and MC piping and components will be governed by ASME Section XI. When applying the criteria of ASME Section XI, one of the considerations is the exemption criteria presented in Paragraphs IWB-1220, IWC-1220, IWD-1220, IWE-1220, IWF-1230 for Class 1, 2, 3 and MC systems respectively. These paragraphs state specific criteria for exemption of components due to various factors including line size, system, function and operation conditions.

A second consideration is the Augmented Inspection Programs that mandates through Regulator requirements, FSAR, TRM, approved relief requests, and/or plant specific criteria.

2.3.6 Updated Final Safety Analysis Report (FSAR) and Technical Requirements Manual (TRM)

In general, the JAF Updated Final Safety Analysis Report (FSAR) provides limited references to inservice inspection requirements. There are numerous references to inservice inspections interspersed throughout the FSAR, but most are inconsequential discussions on accessibility or testing requirements with the exception of the Main Steam and Feedwater Systems and High Stress Break Exclusion. FSAR Chapter 16, Section 16.4 addresses the general requirements of JAF's ISI Program. The JAF FSAR establishes additional inservice inspection commitments which are incorporated into the ISI Program (reference JAF-RPT-03-00289, Augmented Main Steam and Feedwater High Stressed Weld Inspection Program and JAF-SE-03-004, Reduction of Sample Number of Welds Requiring Inspection under the Main Steam and Feedwater Augmented Inspection Program). The TRM also addresses this issue in TRM Surveillance 3.4.A.2 and TRM Bases B 3.4.A.

TABLE 2.3-1 INSERVICE INSPECTION BOUNDARY DRAWINGS	
DRAWING NUMBER	TITLE
ISI-FB-10H	Reactor Building Service Water Cooling System 66
ISI-FB-35E	Control Room Area Service and Chilled Water System 70
ISI–FM-15A	Reactor Building Cooling Water System 15
ISI–FM-15B	Reactor Building Cooling Water System 15
ISI-FM-17A	Radwaste System 20
ISI-FM-18A	Drywell Inerting C.A.D. and Purge System 27
ISI-FM-18B	Drywell Inerting C.A.D Purge and Containment Differential Pressurization System 27
ISI–FM-18C	NUREG 0578 Implementation & Post Accident Sampling System 27
ISI-FM-18D	Containment Hydrogen & Oxygen Sampling System 27
ISI-FM-19A	Fuel Pool Cooling & Clean-Up System 19
ISI-FM-20A	Residual Heat Removal System 20
ISI-FM-20B	Residual Heat Removal System 20
ISI-FM-21A	Standby Liquid Control System 11
ISI-FM-22A	Reactor Core Isolation Cooling System 13
ISI-FM-23A	Core Spray System 14
ISI-FM-24A	Reactor Water Cleanup System 12
ISI-FM-25A	High Pressure Coolant Injection System 23
ISI-FM-26A	Reactor Water Recirculation System 02-2
ISI-FM-27B	Control Rod Drive System 03
ISI-FM-29A	Main Steam System 29
ISI-FM-34A	Feedwater System 34
ISI-FM-39C	Instrument Air Reactor Building & Drywell System 39
ISI-FM-46A	Service Water System 46
ISI-FM-46B	Emergency Service Water System 46 & 15
ISI-FM-47A	Nuclear Boiler Vessel Instruments System 02-3
ISI-FM-48A	Standby Gas Treatment System 01-125
ISI-FM-49A	Drywell/Torus Leak-Rate-Analyzer System 16-1
ISI-FM-93A	Fuel Oil Lines Emergency Diesel Generators System 93
ISI-FM-93C	Engine Cooling & Lubrication Oil Emergency Diesel Generators System 93
ISI-FM-94A	Air Start-Up Lines Emergency Diesel Generators System 93
ISI-FM-119A	Neutron Tip Monitoring System 07 Sheet 1
ISI-FM-119B	Neutron Tip Monitoring System 07 Sheet 2

2.4 ISI Isometric and Component Drawings for Nonexempt ISI Class Components/Supports and Calibration Standards

ISI Isometric and Component Drawings were developed to identify the ISI Class 1, 2, and 3 components (welds, bolting, etc.) and support locations at JAF. These ISI component and support locations are identified on the ISI Isometric and Component Drawings listed in Table 2.4-1. The ISI Class MC components are identified on the CISI Reference Drawings listed in Table 2.4-2. Calibration Standards approved for use at JAF are listed in Table 2.4-3. Additional standards, as approved by JAF, may be designed and fabricated, as needed.

JAF's ISI Program, including the ISI Database, ISI Program Plan, and ISI Selection Document, addresses the nonexempt components, which require examination and testing.

A summary of JAF's ASME Section XI nonexempt components and supports is included in Section 7.0.

Table 2.4-1
JAF Piping Isometric Drawings

ISI Drawing Number	Drawing Title
MSK-3001	Reactor Water Recirculation System
MSK-3002	Reactor Water Recirculation System
MSK-3003	Control Rod Drive System
MSK-3004	Residual Heat Removal System
MSK-3005	Residual Heat Removal System
MSK-3006	Residual Heat Removal System
MSK-3007	Residual Heat Removal System
MSK-3008	Residual Heat Removal System
MSK-3009	Residual Heat Removal System
MSK-3010	Residual Heat Removal System
MSK-3011	Residual Heat Removal System
MSK-3012	Residual Heat Removal System
MSK-3013	Residual Heat Removal System
MSK-3014	Residual Heat Removal System
MSK-3015	Residual Heat Removal System
MSK-3016	Residual Heat Removal System
MSK-3017	Residual Heat Removal System
MSK-3018	Reactor Water Cleanup System
MSK-3019	Reactor Core Injection Cooling
MSK-3020	Reactor Core Injection Cooling
MSK-3021	Core Spray System
MSK-3022	Core Spray System
MSK-3023	Core Spray System
MSK-3024	High Pressure Cooling Injection
MSK-3025	High Pressure Cooling Injection
MSK-3026	High Pressure Cooling Injection
MSK-3027	High Pressure Cooling Injection
MSK-3028	High Pressure Cooling Injection
MSK-3029	Drywell Inerting, C.A.D., and Purge System
MSK-3030	Drywell Inerting, C.A.D., and Purge System
MSK-3031	Main Steam System
MSK-3032	Main Steam System
MSK-3033	Feedwater System
MSK-3034	Feedwater System
MSK-3035	Symbols and Legend for Isometric Drawing in Inservice Inspection (ISI)
MSK-3036	ISI Reactor Vessel Stretch Out For Weld Designation & Insul'n Removal
MSK-3037	RHR Heat Exchanger

MSK-3038	Scam Tank
MSK-3039	Control Drive West Side Piping
MSK-3040	Control Drive East Side Piping
MSK-3116	Jet Pump Assembly

TABLE 2.4-2
CISI REFERENCE DRAWINGS

Drawing	Title
ISI-IWE-001	General Arrangement & Details of Drywell & Torus
ISI-IWE-002	Drywell Shell Stretch out
ISI-IWE-003	Torus Surfaces – North & South Hemispheres
ISI-IWE-004	Typical Containment Penetrations
ISI-IWE-005	Personnel Airlock
ISI-IWE-006	Equipment Door Assembly
ISI-IWE-007	Drywell CRD Hatch
ISI-IWE-008	Drywell Top Head Manway

TABLE 2.4-3
NONDESTRUCTIVE EXAMINATION CALIBRATION STANDARDS

ID No.	Material Type	Size	Thickness	Plant Usage
3-A376-.300	A376 Type 304	3" Sch. 80	0.300" Nominal	CRD
4-A376-.4" 0.337"	A376 Type 304	4" Sch. 80	40.337" Nominal 4"	RECIRC
6-A376-.432	A376 Type 304	6" Sch. 80	0.432" Nominal	RWCU
10-A376-.60	A376 Type 304	10" Sch. 80"	0.594" Nominal	CS
12-A376-.660	A376 Type 304	12" OD x 0.620" wall	0.660" Actual	RECIRC
12-A376-.760	A376 Type 304	12" OD x 0.750" wall	0.760"Actual	RECIRC
20-A376-1.14	A376 Type 304	20" Sch. 80	1.031" Nominal	RHR
22-A376-1.16	A376 Type 304	22" Sch. 80	1.125" Nominal	RECIRC
24-A376-1.25	A376 Type 304	24" Sch. 80	1.218" Nominal	RHR
28-A376-1.53	A376 Type 304	28" OD x 1.400" wall	1.59" Actual	RECIRC
3-A106-.438	A106 GR B	3" Sch. 160	0.438" Nominal	CRD
4-A106-.350	A106 GR B	4" Sch. 80	0.337" Nominal	RHR / RCIC
6-A106-.460	A106 GR B	6" Sch. 80	60.432" Nominal	RWC-UP
6-A106-.864	A106 GR B	6" Sch. XXH	0.864" Nominal	SRV PUP PC
8-A106-.500	A106 GR B	8" Sch. 80	0.500" Nominal	SCRAM DIS
10-A106-.365	A106 GR B	10" Sch. 40	0.365" Nominal	CS
10-A106-.594	A106 GR B	10" Sch.	0.594" Nominal	HPCI
12-A106-.844	A106 GR B	12" Sch. 100	0.844" Nominal	FW
14-A106-.940	A106 GR B	14" Sch. 100	0.938" Nominal	HPCI / FW
18-A106-1.156	A106 GR B	18" Sch. 100	1.156" Nominal	FW
20-A106-.594	A106 GR B	20" Sch. 40	0.594" Nominal	RHR
20-A106-1.05	A106 GR B	20" Sch. 80	1.031" Nominal	RHR
20-A106-1.281	A106 GR B	20" Sch. 100	1.281" Nominal	FW
24-A106-1.32	A106 GR B	24" Sch. 80	1.218"	RHR/MS
PT NO 1	SA533 Class 2 Type B	PLATE	5.00"	CLS. HEAD
PT NO 2	SA533 Class 2 Type B	PLATE	5.00"	BTM. HEAD
PT NO 3	SA533 Class 2 Type B	PLATE	7.00"	RPV LG. / CIR WELD
PT NO 4	SA508 Class 2	PLATE	7.00"	SKT,VES,FLG,LIG
SS-304-12"	A240 Type 304 W / 308L OL	12" Sch. 80	0.688" / 0.600"	RECIR OL / IGSCC
SS-304-22"	A240 Type 304 W / 308L OL	22" Sch. 80	1.125" / 0.500"	RECIR OL / IGSCC
FITZ-RC-INLET-14"NOZ-1	SA312 Type 304 / SA106	F 14" OD x 1.125" wall	1.1" / 1.0" Actual	RECIR INLETNOZ Mockup

TABLE 2.4-3
NONDESTRUCTIVE EXAMINATION CALIBRATION STANDARDS

ID No.	Material Type	Size	Thickness	Plant Usage
FITZ-RC-OUTLET-28"NOZ- 1	SA312F Type 304 / SA106 GR B	28" OD x 1.840" wall	1.6" / 1.8" Actual	RECIR OUTL NOZ Mockup
CRD-CB-NPS-4"	SA508 Class 2 / Inc. 600./ Inc. OL	5.28" OD / 3.860 ID	0.710 / 0.502	CRD NOZZLE OL / IGSCC Mockup
CS-SE-1	A182 F304	13.625	VARYING	CORE SPRAY SE Mockup
JPI-SE-1	SB166 / INCONEL 600	5.28" OD / 3.522 ID	0.879" Actual	JPI NOZ SE Mockup
JPI-SE-2	A182 F304	8" Sch. 80	0.500" Nominal	JPI
M2410144	SA351 GR CF8M / A106 GR B	24" Sch. 100	1.531" Nominal	RHR-Elbow /Valve Mockup
M2410143	SA351 GR CF8M / SA216 WCB	24" Sch. 100	1.531" Nominal	RHR Valve / Valve Mockup
M2410142	A182 F304 / SA216 WCB	24" Sch. 100	1.531" Nominal	RHR Valve / Tee Mockup
M2802230	A403 WP304	28" OD x 24" ID	2.0" Nominal	RECIRC Cross to Tee Mockup
M2802285	A403 WP304	28" OD x 24" ID 2.0"	2.0" Nominal	RECIRC Cross to Reducer Mockup
DECON-1	A376 F304	4" OD x 1.200" Wall	1.2" Nominal	RECIRC DECN FLG
JAF-85	A376 F304	4" Sch. 80	0.337" Nominal	JPI
JAF-86	A376 F304	12" Sch. 80	0.688" Nominal	JPI
MSK-3095	A106 GR B	14" Sch. 80 0	0.750" Nominal	HPCI
MSK-3094	A106 GR B	24" Sch. 40	0.688" Nominal	RHR
OLB-10-SS-.550	A312 Type 304 / OL	10" Sch. 80"	" 0.594" / 0.300"	CS WOL
RL-1 SA240	SA240 Type 304	PLATE	RL=.1-.3	SS Sizing Block
RL-2 SA240	SA240 Type 304	PLATE	RL=.4 - .7	SS Sizing Block
RL-3 SA240	SA240 Type 304	PLATE	RL=.8-1.0	SS Sizing Block
RL-4 SA240	SA240 Type 304	PLATE	RL=1.1-1.3	SS Sizing Block
RL-5 SA240	SA240 Type 304	PLATE	RL=1.4-1.6	SS Sizing Block
RL-6 SA240	SA240 Type 304	PLATE	RL=1.7-1.9	SS Sizing Block
10-A403-.600	A403 Type 347 / A182 Type 304	10" OD x 0.600" Wall	0.600" Actual	CS "B" Loop
12-A106-.375	SA106 GR B	12" Sch. Std.	0.375" Nominal	RHR

TABLE 2.4-3
NONDESTRUCTIVE EXAMINATION CALIBRATION STANDARDS

ID No.	Material Type	Size	Thickness	Plant Usage
24-A106-.375	SA106 GR B	24" Sch. Std.	. 0.375" Nominal	RHR
B907	SA540 B24 / SA540 B23	6"	48.875	RPV Stud
2558	A276 Type 304	PLATE	1.50”	RPV CLAD SIZING
CB-09-40	SA508 Class 2	9" OD x 1.688” Wall	1.688" Actual	RPV Nozzle
CB-08-37	A106 GR B	8" Sch. Std.	0.322" Nominal	Core Spray
CB-01-38	SA516 Gr. 70	Plate	CB-01-38 SA516 Gr. 70 Plate 1.125	RHR Hx
CB-01-39	SA516 Gr. 70	Plate	CB-01-39 SA516 Gr. 70 Plate 0.875 RHR Hx	RHR Hx
10853	Carbon Steel	Carbon Steel Plate	0.500” – 2.000”	PDI Alternate Calibration Block
10854	Type 304 SS	Plate	0.500” – 2.000”	PDI Alternate Calibration Block
10855	Type 316 SS	Plate	0.500” – 2.000”	PDI Alternate Calibration Block
CB-02-212	A106 Gr B	2” Sch 80	0.218” Nominal	Small Bore Piping ASME Calibration Blocks
CB-02-213	A312 TP 304 SS	2” Sch 80	0.218” Nominal	Small Bore Piping ASME Calibration Blocks
CB-02-214	A106 Gr B	2” Sch 160	0.344” Nominal	Small Bore Piping ASME Calibration Blocks
CB-02-215	A312 TP 304 SS	1.5” Sch 80	0.200” Nominal	Small Bore Piping ASME Calibration Blocks

2.5 Technical Approach and Positions

Where the requirements of ASME Section XI are not easily interpreted, JAF has reviewed general licensing/regulatory requirements and industry practice to determine a practical method of implementing the Code requirements. The Technical Approach and Position (TAP) documents contained in this section have been provided to clarify JAF's implementation of ASME Section XI requirements. An index which summarizes each TAP is included in Table 2.5-1.

TABLE 2.5-1
TECHNICAL APPROACH AND POSITIONS INDEX

Technical Approach and Position Number	Revision Date[1]	Status[2]	Description of Technical Approach and Position[3]
Reserved			

Note 1: The revision listed is the latest revision of the Technical Approach and Position. The date this revision became effective is the date of the JAF Plant Staff approval of the ASME Section XI interpretation.

Note 2: This column represents the current status of the latest revision.

Note 3: This column includes a description of the Technical Approach and Position on the ASME Section XI requirements and the JAF interpretation.

TECHNICAL APPROACH AND POSITION NUMBER XX.XX.XXXX
Revision 0

COMPONENT IDENTIFICATION:

Code Class:	RESERVED
Reference:	
Examination Category:	
Item Number:	
Description:	
Component Number:	

CODE REQUIREMENT:

POSITION:

3.0 COMPONENT ISI PLAN

The JAF Component ISI Plan includes ASME Section XI nonexempt pressure retaining welds, piping structural elements, pressure retaining bolting, attachment welds, pump casings, valve bodies, reactor vessel interior, reactor vessel interior attachments, and reactor vessel core support structures of ISI Class 1, 2, and 3 components that meet the criteria of Subarticle IWA-1300. These components are identified on the ASME Section XI ISI Drawings listed in Section 2.3, Table 2.3-1. Procedure ER-AA-330-002, "Inservice Inspection of Section XI Welds and Components", implements the ASME Section XI Welds and Components ISI Plan. This Component ISI Plan also includes augmented inspection program requirements specified by documents other than ASME Section XI as referenced in Section 2.2 of this document.

3.1 Nonexempt ISI Class Components

The ISI Class 1 nonexempt components subject to examination are those that are not exempted under the criteria of Paragraph IWB-1220 in the 2007 Edition through the 2008 Addenda. The JAF ISI Class 2 and 3 nonexempt components identified on ISI Drawings are those not exempted under the criteria of Paragraphs IWC-1220 and IWD-1220 in the 2007 Edition through the 2008 Addenda of ASME Section XI. A summary of JAF ASME Section XI nonexempt components is included in Section 7.0.

3.1.1 Identification of ISI Class 1, 2, and 3 Nonexempt Components

ISI Class 1, 2, and 3 nonexempt components are identified on the ISI Isometric (Weld Identification) and ISI Component Drawings listed in Section 2.4, Table 2.4-1. Welded attachments are also identified by controlled JAF individual support detail drawings.

3.1.2 Components Exempt from Examination

Certain components or parts of components may be exempted from examination based on design and accessibility per the requirements of Paragraphs IWB-1220, IWC-1220, and IWD-1220.

The process for exempting JAF components from the Component ISI Plan per Paragraphs IWB-1220, IWC-1220, and IWD-1220 is included in the ISI Database. These sections include discussions of exempt components and the bases for those exemptions.

3.2 Risk-Informed Examination Requirements

Piping welds that fall under RI-ISI Examination Category R-A are classified as either High Safety Significant (HSS) or Low Safety Significant (LSS). Per ASME Code Case N-716-1, piping welds classified as HSS are subject to examination while piping welds classified as LSS are not subject to examinations (except for pressure testing). Thin wall welds that were excluded from volumetric examination under ASME Section XI rules per Table IWC-2500-1 are included in the scope that is potentially subject to RI-ISI examination at JAF. Class 2 components, excluding attachment welds and supports are also classified as either HSS or LSS.

Piping welds that are inspected for cause under certain other JAF programs such as the Flow Accelerated Corrosion (FAC) or IGSCC Programs are still included in the total population where selections are made. Low-Safety Significant Piping Welds are assigned Item No. R0.00. Welds subject to a degradation method of Thermal Fatigue are assigned Item No. R1.11. Welds subject to a degradation method of Erosion Cavitation are assigned Item No. R1.13. Welds subject to a degradation method of Crevice Corrosion Cracking are assigned Item No. R1.14. Welds subject to a degradation method of Primary Water Stress Corrosion Cracking (PWSCC) are assigned Item No. R1.15. Welds subject to a degradation mechanism of Intergranular or Transgranular Stress Corrosion Cracking (IGSCC or TGSCC) are assigned Item No. R1.16. Welds subject to a degradation mechanism of localized corrosion [Microbiologically Influenced Corrosion (MIC) or Pitting] are assigned Item No. R1.17. Welds subject to a degradation mechanism of Flow Accelerated Corrosion (FAC) only are assigned Item No. R1.18. Welds subject to a degradation mechanism of External Chloride Stress Corrosion Cracking (ECSCC) are assigned Item No. R1.19. Welds not subject to a degradation method mechanism are assigned Item No. R1.20.

All welds classified as HSS are included in the total population which is used for meeting the selection criteria of Code Case N-716-1.

The Item Numbers associated with R-A are provided in Table 3.2-1.

TABLE 3.2-1
Risk Informed Examination Categories

Item Number	Parts Examined
R1.11	Welds Subject to Thermal Fatigue
R1.13	Welds Subject to Erosion-Cavitation
R1.14	Welds Subject to Crevice Corrosion Cracking
R1.16	Welds Subject to Intergranular or Transgranular Stress Corrosion Cracking (IGSCC or TGSCC)
R1.17	Welds Subject to Localized Corrosion (MIC/Pitting)
R1.18	Welds Subject to Flow Accelerated Corrosion (FAC)
R1.20	Welds Not Subject to a Degradation Mechanism

3.3 ISI Class 1 Piping Size Exemption for Water and Steam

As stated above, an exemption from the surface and volumetric examination requirements of Subarticle IWB-2500 is available, provided that site specific calculations are prepared to determine the size of ISI Class 1 water and steam piping and components that would fully satisfy the makeup criteria of Paragraph IWB-1220(a) from the 2007 Edition through the 2008 Addenda of ASME Section XI.

JAF has decided not to perform this calculation, therefore the makeup exemption of Paragraph IWB-1220(a) is not invoked for ISI Class 1 piping and components at JAF.

4.0 SUPPORT ISI PLAN

The JAF Support ISI Plan includes the supports of ASME Section XI nonexempt ISI Class 1, 2, and 3 components as described in Section 3.0; and ISI Class MC components as described in Section 6.0. Procedure ER-AA-330-003 "Inservice Inspection of Section XI Component Supports", implements the ASME Section XI Support ISI Plan.

4.1 Nonexempt ISI Class Supports

The JAF ISI Class 1, 2, 3, and MC nonexempt supports are those which do not meet the exemption criteria of Paragraph IWF-1230 of ASME Section XI. A summary of JAF ASME Section XI nonexempt supports is included in Section 7.0.

4.1.1 Identification of ISI Class 1, 2, and 3 Nonexempt Supports

ISI Class 1, 2, and 3 supports are identified on the ISI Isometrics and Component Drawings listed in Section 2.4, Table 2.4-1. Supports are identified by controlled JAF individual support detail drawings.

ISI Class MC supports are identified on the JAF CISI Drawings listed in Section 2.4, Table 2.4-2.

4.2 Snubber Examination and Testing Requirements

4.2.1 The 2006 Addenda of ASME Section XI deleted the requirement for examination of snubbers and deleted reference to the OM Code subsection ISTD for snubber testing. Section XI retained the requirements for examination of the support containing the snubber, but the examination of the snubber itself is now solely under the OM Code.

The ASME Section XI ISI Program uses Subsection IWF to define support inspection requirements. The ISI Program maintains the Code Class snubbers in the populations subject to inspection per Article IWF-2000.

4.2.2 ASME Section XI Paragraph IWF-1300 requires integral and non-integral attachments for snubbers to be examined in accordance with Subsection IWF of ASME Section XI. This results in VT-3 visual examination of the snubber attachment hardware including the bolting, pins, and their interface to the clamp, but does not include the component-to-clamp interface.

The ASME Section XI ISI Program uses Subsection IWF to define the inspection requirements for all ISI Class 1, 2, and 3 supports, regardless of type. The ISI Program maintains the Code Class snubbers in the support populations subject to inspection per Article IWF-2000. This is done to facilitate scheduling and inspection requirements of the snubber attachment hardware (e.g., bolting, pins and their interface to the clamp).

It should be noted that the examination of snubber welded attachments will be performed in accordance with the ASME Section XI Subsections IWB, IWC, and

IWD welded attachment examination requirements (e.g.; Examination Categories B-K, C-C, and D-A).

5.0 SYSTEM PRESSURE TESTING ISI PLAN

The JAF System Pressure Testing ISI Plan includes ASME Section XI nonexempt ISI Class 1, 2, 3, and 3A (Augmented) systems. Procedure ER-AA-330-001 "Section XI Pressure Testing", implements the ASME Section XI Pressure Testing ISI Plan.

5.1 Nonexempt ISI Class Systems

5.1.1 Identification of ISI Class 1, 2, 3 and 3A (Augmented) Systems

ISI Class 1, 2, 3, and 3A (Augmented) nonexempt systems are identified on the ISI Isometric (Weld Identification), ISI Component Drawings and ISI Boundary Drawings listed in Section 2.4, Table 2.3-1 and Table 2.4-1.

5.1.1.1 Augmented Systems

Systems for the pressure test program are discussed in Engineering Report JAF-RPT-MISC-00658.

5.1.1.1.1 Standby Liquid Control System (SLC)

Based on the operational parameters of the Class 1 portion of the Standby Liquid Control System (SLC), which is not exposed to boron, the requirements of IWA-5241(f), (g), and (h) shall not be applied.

5.1.1.1.2 Containment Penetrations

For containment penetration configurations, where there are no inside valves, the following will apply:

- The rules of Section XI shall apply to the first valve outside the drywell through the penetration to the first weld inside the drywell.
- Repairs/Replacements shall be in accordance with JAF's design specifications and the original construction code.
- Pressure tests shall be in accordance with the Appendix J Program.

5.1.1.1.3 Class 2 Control Rod Drive System (CRD)

All pressure tests shall be performed during Reactor startup and/or shutdown during a refueling and/or maintenance outage. Reference CRD testing requirements, memorandum PEP-RAP-99-025, dated February 9, 1999, "ISI Pressure Testing of the CRD System" and JAF-ICD-CRD-04093, Rev. 0.

5.1.1.1.4 Emergency Diesel Generator (EDG)

JAF has optionally upgraded portions of the EDG system to an augmented Class 3 Category (3A) for the purposes of pressure testing only.

5.1.2 Identification of System Pressure Tests

JAF System Pressure Tests are identified in the Operations Department Surveillance Test Procedures and the Pressure Test Matrix. Pressure tests are scheduled using the PM process.

The Site Surveillance Testing Procedures are listed below in Table 5.1-1.

Table 5.1-1: Site Surveillance Testing Procedures	
ST-20U	CRD Class 2 Piping Inservice Leakage Test (ISI)
ST-3NA	Core Spray Loop A Class 2 Piping Leakage Test (ISI)
ST-3NB	Core Spray Loop B Class 2 Piping Leakage Test (ISI)
ST-3T	Core Spray Class 1 Piping System Leakage Test for 10-Year Inspection Interval (ISI)
ST-8K	Control Room Service and Refrigeration Water Chiller Class 3 Piping System Leakage Test (ISI)
ST-8L	Emergency Service Water, Reactor Building Cooling and Service Water Class 3 Piping System Leakage Test (ISI)
ST-9FA	EDG A and C Air Start Class 3 Piping Leakage Test (ISI)
ST-9FB	EDG B and D Air Start Class 3 Piping Leakage Test (ISI)
ST-9GA	EDG A and C Fuel Oil, Combustion Air and Exhaust Class 3 Piping Leakage Test (ISI)
ST-9GB	EDG B and D Fuel Oil, Combustion Air and Exhaust Class 3 Piping Leakage Test (ISI)
ST-9HA	EDG A and C Lube Oil and Cooling Water Systems Class 3 Piping Leakage Test (ISI)
ST-9HB	EDG B and D Lube Oil and Cooling Water Systems Class 3 Piping Leakage Test (ISI)
ST-41E	Fuel Pool Cooling Class 3 Piping System Leakage Test (ISI)
ST-4L	HPCI Class 2 Piping Functional Test (ISI)
ST-24H	RCIC Class 2 and 3 Piping System Leakage Test (ISI)
ST-2AK	RHR Loop B Containment Spray Headers and Nozzles Air Test (ISI)
ST-2UA	RHRSW Loop A Class 3 Piping System Leakage Test (ISI)
ST-2UB	RHRSW Loop B Class 3 Piping System Leakage Test (ISI)
ST-2AJ	RHR Loop A Containment Spray Headers and Nozzles Air Test (ISI)
ST-2WA	RHR Loop A Keep-Full and Torus Cooling Class 2 Piping Leakage Test (ISI)
ST-2WB	RHR Loop B Keep-Full and Torus Cooling Class 2 Piping Leakage Test (ISI)
ST-2ZA	RHR Loop A Shutdown Cooling Class 2 Piping Leakage Test (ISI)

ST-2ZB	RHR Loop B Shutdown Cooling Class 2 Piping Leakage Test (ISI)
ST-2AI	RHR Class 1 Piping System Leakage Test for 10-Year Inspection Interval (ISI)
ST-39K	Shutdown Cooling Suction Piping System Leakage Test for 10-Year Inspection Interval (ISI)
ST-39H	RPV System Leakage Test and CRD Class 2 Piping Inservice Test (ISI)
ST-39HA	RPV System Leakage Test and CRD Class 2 Piping Pressure Test (10 year)
ST-6KA	A SLC System Class 2 Piping Leakage Test (ISI) and Operability Test (IST)
ST-6KB	B SLC System Class 2 Piping Leakage Test (ISI) and Operability Test (IST)
ST-6M	Standby Liquid Control Recirculation, Injection Test (IST, ISI)

5.1.2.1 Repair/Replacement Activity System Pressure Tests

A system pressure test conducted following a repair or replacement performed per AP-19.02. When repair or replacement tests are coincident with the periodic system pressure test, additional requirements that may be imposed by AP-19.02 must be considered.

5.2 Special Instructions

5.2.1 JAF is required to meet the system pressure test requirements of ASME Section XI 2007 Edition, 2008 Addenda as required by 10CFR50.55a

5.2.2 The pressure retaining components within the ASME Section XI Class Boundary of each system shall be subjected to system pressure tests at which time a VT-2 examination performed to detect leakage.

5.2.3 Routine (outage, period, or interval) system pressure tests shall be performed per Surveillance Test procedures (STs). As a minimum, routine pressure test procedures and test criteria shall comply with ASME Boiler and Pressure Vessel Code, Section XI 2007 Edition, 2008 Addenda and the ISI Program Plan.

5.2.4 Relief from ASME Section XI requirements will be accomplished by submittal of relief requests to the NRC and approval by NRC as required by 10CFR50.55a.

5.2.5 The Site Materials Enginer shall review Surveillance Test procedures that affect this test program.

5.2.6 The system test conditions (i.e., pressure and temperature) shall be maintained essentially constant during the course of the visual examination unless provisions of elevated temperatures are evoked (2007 edition through 2008 addenda, IWA-5245).

5.2.7 For test pressurization boundaries, due to the possible combinations of Code Editions and Code Cases, the most stringent test boundaries specified in an applicable Code Case or Code Edition shall be used.

5.2.8 Where the system hydrostatic test following a component repair or replacement would impose system conditions that conflict with limitations included in Tech Specs, a system leakage test [2007 edition through 2008 addenda, IWA-5211(a)] at nominal operating temperature shall be acceptable.

5.2.9 Leakage from components associated with A/B RHR Systems, A/B Core Spray Systems, HPCI, RWR, SBGT and RWCU shall be immediately documented by initiation of a WO per PI-AA-120 and WC-AA-106.

5.3 General Requirements

5.3.1 System periodic pressure testing for ASME Class 1 systems is performed on a refueling outage and 10 year interval basis.

5.3.1.2 System leakage tests are required following each refueling outage and system hydrostatic tests are required once per 10 year interval.

5.3.1.3 The system hydrostatic test satisfies system leakage test requirements if performed.

5.3.2 System periodic pressure testing for ASME Class 2 and 3 systems is performed on a 40 month period and 10 year interval basis.

5.3.2.1 System pressure tests are required once per 40 month inspection period and system hydrostatic tests are required once per 10 year interval.

5.3.2.2 The system hydrostatic test satisfied the system pressure test requirements if performed.

5.3.3 System periodic pressure testing is performed per ASME Section XI 2007 Edition, 2008 addenda unless Code Cases are applied in lieu of the required pressure testing.

5.3.4 Pressure testing following welded repairs and replacements is performed per AP-19.02, however periodic system pressure tests can be used to satisfy the required post work pressure tests.

5.3.5 Reference values for design pressures, design temperatures, and operating pressures may be taken from the JAF Line Designation Table, from construction/preop hydrostatic test records, or from engineering modification documents.

5.3.5.1 Testing of the Reactor Pressure Vessel shall be per Tech Spec limitations for temperature, pressure, heat-up, and cool down rates.

5.3.6 When performing system hydrostatic tests [IWA-5211(b)] per requirements of ASME Section XI Edition 2007/08 the instruments for pressure tests shall meet the requirements of IWA-5260.

5.3.7 Pressure testing of augmented classified systems shall be in accordance with the ISI class number assigned (i.e., Class 3A shall be tested per Class 3 requirements, etc.).

5.4 References

5.4.1 AP-19.02, Post Work Pressure Testing and Visual Inspection Requirements
5.4.2 QAPM, Exelon Quality Assurance Program Manual
5.4.3 10CFR50.55a, Codes and Standards
5.4.4 [TRM], Technical Requirements Manual Section 3.4, Reactor Coolant Systems (RCS)
5.4.5 NRC Regulatory Guide 1.26, Quality Group Classification for Water, Steam and Radioactive Waste containing Components for Nuclear Power Plants
5.4.6 NRC Generic Letter 90-05, Guidance for Performing Temporary Non-Code Repair of ASME (ISI) Code Class 1, 2, 3, Piping
5.4.7 NRC NUREG-0737, Clarification of TMI Action Plan No. III.D.1, Primary Coolant Sources Outside the Containment

6.0 CONTAINMENT ISI PLAN

The JAF Containment ISI Plan includes ASME Section XI ISI Class MC pressure retaining components and their welded attachments. The Containment ISI Plan also includes information related to augmented examination areas, component accessibility, and examination review.

6.1 Nonexempt ISI Class Components

A summary of JAF ASME Section XI nonexempt CISI components is included in Section 7.0.

The process for scoping JAF components for inclusion in the Containment ISI Plan is included in the containment sections of the ISI Database. These sections include a listing and detailed basis for inclusion of containment components.

Components that are classified as ISI Class MC, must meet the requirements of ASME Section XI in accordance with 10 CFR 50.55a(g)(4). Although supports of IWE components are not strictly required to be examined in accordance with 10 CFR 50.55a(g)(4)(v), JAF has elected to perform these examinations in accordance with ASME Section XI, Subsection IWF.

6.1.1 Identification of ISI Class MC Nonexempt Components

ISI Class MC components are identified on the JAF CISI Drawings listed in Section 2.4, Table 2.4-2. These drawings are identified with special prefixes of "ISI-IWE-" at JAF.

6.1.2 Identification of ISI Class MC Exempt Components

Certain containment components or parts of components may be exempted from examination based on design and accessibility per the requirements of Paragraph IWE-1220.

The process for exempting JAF components from the Containment ISI Plan per Paragraph IWE-1220 is included in the containment sections of the ISI Database. These sections include a listing and basis for exempting applicable components.

6.2 Augmented Examination Areas

Metal containment surface areas subject to accelerated degradation and aging require augmented examination per Examination Category E-C and Paragraph IWE-1240.

No areas requiring augmented examinations under IWE-1240 were identified in the First CISI Interval. Areas were identified in the Second CISI Interval as requiring application of additional augmented examination requirements under Paragraph IWE-1240 as a result of license renewal.

6.2.1 Ultrasonic thickness measurements shall be performed from the exterior surface of the Torus Shell in accordance with JAF calculation JAF-CALC-16-00008, Rev. 0, JAF Torus Corrosion Allowance. These examinations are being performed in support of the Torus Preservation Program. These examinations will be performed on a periodic basis (see 2.2.5).

6.2.2 JAF does not have a moisture barrier at the interface of the concrete to steel shell area. IWE-1241 requires augmented examinations of this area because it is subject to accelerated degradation and aging. These examinations will be performed on a periodic basis (see 2.2.17).

6.3 Component Accessibility

ISI Class MC components subject to examination shall remain accessible for either direct or remote visual examination from at least one side for the life of the plant per the requirements of ASME Section XI, Paragraph IWE-1231.

Paragraph IWE-1231(a) requires as a minimum, the following portions of Class MC containment vessels, parts and appurtenances, and Class CC metallic shell and penetration liners shall remain accessible for either direct or remote visual examination, from at least one side of the vessel, for the life of the plant:

1) openings and penetrations;

2) structural discontinuities;

3) 80% of the pressure-retaining boundary (excluding attachments, structural reinforcement, and areas made inaccessible during construction) and

4) surface areas identified in IWE-1240

(b) The requirements of IWE-1232 shall be met when accessibility for visual examination is only from the interior surface.

Portions of components embedded in concrete or otherwise made inaccessible during construction are exempted from examination, provided that the requirements of ASME Section XI, Paragraph IWE-1232 have been fully satisfied.

In addition, inaccessible surface areas exempted from examination include those surface areas where visual access by line of sight with adequate lighting from permanent vantage points is obstructed by permanent plant structures, equipment, or components; provided these surface areas do not require examination in accordance with the inspection plan, or augmented examination in accordance with Paragraph IWE-1240.

6.4 Responsible Individual

ASME Section XI Subsection IWE-2320 requires a Responsible Individual to be involved in the development of plans and procedures for general visual examination of containment surfaces, instruction, training, and approval of general visual examination personnel, performance or direction of general visual examinations and evaluation of general visual examination results and documentation.

7.0 COMPONENT SUMMARY TABLES

7.1 Inservice Inspection Summary Tables

The following Table 7.1-1 provides a summary of the ASME Section XI pressure retaining components, supports, containment structures, system pressure testing, and augmented inspection program components for the Fifth ISI Interval and Third CISI Interval at JAF.

The format of the Inservice Inspection Summary Tables is as depicted below and provides the following information:

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System	Relief Request/ Technical Approach & Position Number	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)

(1) Examination Category (with Examination Category Description):

Provides the Examination Category and description as identified in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, and IWF-2500-1. Only those Examination Categories applicable to JAF are identified.

Examination Category "R-A" from ASME Code Case N-716-1 is used in lieu of ASME Section XI Examination Categories B-F, B-J, C-F-1and C-F-2 to identify ISI Class 1 and Class 2 piping welds for the RI-ISI Program. In addition, Categories C-A and C-B Class 2 components are incorporated into the implementation of Code Case N-716-1.

(2) Item Number (or Augmented Number):

Provides the Item Number as identified in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, and IWF-2500-1. Only those Item Numbers applicable to JAF are identified.

For piping welds under the RI-ISI Program, the RI-ISI Item Number based on the degradation mechanism assigned is provided in this column of the table.

Specific abbreviations such as reference paragraph numbers (2.2.1, 2.2.2 and 2.2.3) have been developed to identify Augmented Inspection Programs and other JAF commitments.

(3) Description:

Provides the description as identified in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, and IWF-2500-1.

For Risk-Informed piping welds, a description of the RI-ISI Item Number is provided.

For Augmented Inspection Programs, a description of the augmented basis is provided.

(4) Examination Requirements:

Provides the examination methods required by ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, and IWF-2500-1.

Provides the examination requirements for piping welds under the RI-ISI Program that are in accordance with the ASME Code Case N-716-1.

Provides the examination requirements for Augmented Inspection Program components.

(5) Total Number of Components by System:

Provides the system designator (abbreviations). Reference Table below:

System	Sub Sys	Description
02		Reactor Pressure Vessel (RPV)
2	2	Reactor Water Recirculation (RC)
2	3	Nuclear Boiler Vessel Inst. (NBI)
3		Control Rod Drive (CRD)
10		Residual Heat Removal (RHR)
11		Standby Liquid Control (SLC)
12		Reactor Water Cleanup (RWC)
13		Reactor Core Isolation Cooling (RCIC)
14		Core Spray (CS)
15		Reactor Bldg Closed Loop Cooling Water (RBCLC)
16		Primary Containment (PC)

19	1	Fuel Pool Cooling and Cleanup (FPC)
23		High Pressure Coolant Injection (HPCI)
27		Drywell Inerting CAD & Purge
29		Main Steam (MS)
34		Feedwater (FW)
46		Service Water System (SWS)
66		Reactor Building Vent and Cooling (RBVC)
66		Control Room Vent and Cooling (CRV)

This column also provides the number of components within a particular system for that ASME Section XI Item Number, RI-ISI Item Number and, or Augmented Number.

Note that the total numbers of components by system are subject to change after completion of plant modifications, design changes, and ISI system classification updates, and will be maintained only within the ISI database.

(6) Relief Request/Technical Approach & Position Number:

Provides a listing of Relief Request/TAP Numbers applicable to specific components, the ASME Section XI Item Number, RI-ISI Item Number and Risk Category Number, or Augmented Number. Relief Requests and TAPs that generically apply to all components, or an entire class are not listed. If a Relief Request/TAP Number is identified, reference the corresponding relief request in Section 8.0 or the TAP Number in Section 2.5.

(7) Notes:

Provides a listing of program notes applicable to the ASME Section XI Item Number, RI-ISI Item Number and Risk Category Number, or Augmented Number. If a program note number is identified, reference the corresponding program note in Table 7.1-2.

TABLE 7.1-1
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System	Relief Request/ Technical Approach & Position Number	Notes
B-A Pressure Retaining Welds in Reactor Vessel	B1.11	Circumferential Shell Welds (Reactor Vessel)	Volumetric	RPV: 4	RR-19	19
	B1.12	Longitudinal Shell Welds (Reactor Vessel)	Volumetric	RPV: 12		
	B1.21	Circumferential Head Welds (Reactor Vessel)	Volumetric	RPV: 3		
	B1.22	Meridional Head Welds (Reactor Vessel)	Volumetric	RPV: 22		
	B1.30	Shell-to-Flange Weld (Reactor Vessel)	Volumetric	RPV: 1		
	B1.40	Head-to-Flange Weld (Reactor Vessel)	Volumetric & Surface	RPV: 1		20
B-D Full Penetration Welded Nozzles in Vessels	B3.90	Nozzle-to-Vessel Welds (Reactor Vessel)	Volumetric	RPV: 28	JAF-I5R-02	13, 14
	B3.100	Nozzle Inside Radius Section (Reactor Vessel)	Volumetric	RPV: 28	JAF-I5R-02	13, 15

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System	Relief Request/ Technical Approach & Position Number	Notes
B-G-1 Pressure Retaining Boling, Greater Than 2 in. (50 mm) In Diameter	B6.10	Closure Head Nuts (Reactor Vessel)	Visual, VT-1	RPV: 1 (52 Nuts)		
	B6.20	Closure Studs (Reactor Vessel)	Volumetric	RPV: 1 (52 Studs) RC: 2 (32 Nuts, Washers)		
	B6.40	Threads in Flange (Reactor Vessel)	Volumetric	RPV: 1 (52 Threads)	RR-JAF-I5R-04	
	B6.50	Closure Washers, Bushings (Reactor Vessel)	Visual, VT-1	RPV: 1 (52 Washers) RPV: 1 (52 Bushings)		
	B6.180	Bolts and Studs (Pumps)	Volumetric	RC: 2 (32 Bolts, Studs)		
	B6.190	Flange Surface, when connection disassembled (Pumps)	Visual, VT-1	RC: 2		
	B6.200	Nuts, Bushings, and Washers (Pumps)	Visual, VT-1	RC: 2 (32 Nuts, Bushings, Washers)		
B-G-2 Pressure Retaining Bolting, 2 in. (50 mm) and Less In Diameter	B7.10	Bolts, Studs, and Nuts (Reactor Vessel)	Visual, VT-1	RPV: 3		
	B7.50	Bolts, Studs, and Nuts (Piping)	Visual, VT-1	RC: 2 MS: 11		
	B7.60	Bolts, Studs, and Nuts (Pumps)	Visual, VT-1	RC: 4		
	B7.70	Bolts, Studs, and Nuts (Valves)	Visual, VT-1	RC: 4 RHR: 2 CS: 2 MS: 30 FW: 2		

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System	Relief Request/ Technical Approach & Position Number	Notes
B-K Welded Attachments for Vessels, Piping, Pumps and Valves	B10.10	Welded Attachments (Pressure Vessels)	Surface or Volumetric	RPV: 5		
	B10.20	Welded Attachments (Piping)	Surface	CS: 9 FW: 18 HPCI: 5 MS: 38 RC: 18 RCIC: 21 RHR: 17 RWC: 7		
	B10.30	Welded Attachments (Pumps)	Surface	RC: 6		
B-L-2 Pump Casings	B12.20	Pump Casings (Pumps)	Visual, VT-3	RC: 2		
B-M-2 Valve Bodies	B12.50	Valve Bodies (Exceeding NPS 4 (DN 100) (Valves)	Visual, VT-3	CS: 8 FW: 6 HPCI: 4 MS:19 RC: 4 RCIC: 2 RHR: 11 RWC: 4		

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System	Relief Request/ Technical Approach & Position Number	Notes
B-N-1 Interior of Reactor Vessel	B13.10	Vessel Interior (Reactor Vessel)	Visual, VT-3		RR-JAF-I5R-02	11
B-N-2 Welded Core Support Structures and Inerior Attachments to Reactor Vessels (CM-9 & CM-10)	B13.20	Interior Attachments Within Beltline Region (Reactor Vessel)	Visual, VT-1		RR-JAF-I5R-02	11
	B13.30	Interior Attachments Beyond Beltline Region (Reactor Vessel)	Visual, VT-3		RR-JAF-I5R-02	11
	B13.40	Core Support Structure (Reactor Vessel)	Visual, VT-3		RR-JAF-I5R-02	11
B-O Pressure Retaining Welds in Control Rod Drive and Instrument Nozzle Housings	B14.10	Welds in Control Rod Drive (CRD) Housing (Reactor Vessel) (10% of Peripheral CRD Housings)	Volumetric or Surface	RPV: (56 CRD Housings are peripheral. Examine 6)		9

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System	Relief Request/ Technical Approach & Position Number	Notes
B-P All Pressure Retaining Components	B15.10	System Leakage Test (IWB-5220)	Visual, VT-2	Class 1 Piping		
	B15.20	System Leakage Test (IWB-5220)	Visual, VT-2	Class 1 Piping		

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System	Relief Request/ Technical Approach & Position Number	Notes
C-A Pressure Retaining Welds in Pressure Vessels	C1.10	Shell Circumferential Welds (Pressure Vessels)	Volumetric	RHR: 4		7
	C1.20	Head Circumferential Welds (Pressure Vessels)	Volumetric	CRD: 4 RHR: 2		7
C-B Pressure Retaining Nozzle Welds in Vessels	C2.21	Nozzles Without Reinforcing Plate in Vessels, Greater Than 1/2" (13 mm) Nominal Thickness Nozzle-to-Shell (Nozzle to Head or Nozzle to Nozzle) Weld (Pressure Vessels)	Volumetric & Surface	CRD: 2 RHR: 4		7
	C2.22	Nozzles Without Reinforcing Plate in Vessels Greater Than 1/2” (13 mm) Nominal Thickness Nozzle Inside Radius Section (Pressure Vessels)	Volumetric	CRD: 2 RHR: 4		7
C-C Welded Attachments for Vessels, Piping, Pumps and Valves	C3.10	Welded Attachments (Pressure Vessels)	Surface	CRD: 8 RHR: 12		
	C3.20	Welded Attachments (Piping)	Surface	CS: 27 HPCI: 27 MS: 8 RCIC: 1 RHR: 142		
	C3.30	Welded Attachments (Pumps)	Surface	CS: 2 RHR: 4		

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System	Relief Request/ Technical Approach & Position Number	Notes
C-H All Pressure Retaining Components	C7.10	System Leakage Test (IWC-5220)	Visual, VT-2	Class 2 Piping	RR-JAF-I5R-03	

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System	Relief Request/ Technical Approach & Position Number	Notes
D-A Welded Attachments for Vessels, Piping, Pumps, and Valves	D1.20	Welded Attachments (Piping)	Visual, VT-1	CRV: 10 FPC: 16 RBCLC: 41 RBVC: 2 RCIC: 10 RHR: 44 SWS: 31		
	D1.30 D.140	Welded Attachments (Pumps) Welded Attachments (Valves)	Visual, VT-1 Visual, VT-1	RHR: 8 SWS: 2		
D-B All Pressure Retaining Components	D2.10	System Leakage Test (IWD-5220)	Visual, VT-2	Class 3 Piping		

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components	Relief Request/ Technical Approach & Position Number	Notes
E-A Containment Surfaces	E1.11	Containment Vessel Pressure Retaining Boundary - Accessible Surface Areas	General Visual	14		

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components	Relief Request/ Technical Approach & Position Number	Notes
	E1.12	Containment Vessel Pressure Retaining Boundary - Wetted Surfaces of Submerged Areas	Visual, VT-3	1		
	E1.20	Containment Vessel Pressure Retaining Boundary - BWR Vent System Accessible Surface Areas	Visual, VT-3	2		
	E1.30	Moisture Barriers	General Visual	0		
E-C Containment Surfaces Requiring Augmented Examination	E4.11	Containment Surface Areas - Visible Surfaces	Visual, VT-1	1		
	E4.12	Containment Surface Areas - Surface Area Grid Minimum Wall Thickness Location	Ultrasonic Thickness	1		12
E-G Pressure Retaining Bolting	E8.10	Bolted Connections	Visual, VT-1	23		5

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System						Relief Request/ Technical Approach & Position Number	Notes
				A One-Directional	B Multi-Direction	C Thermal Movement	D Sliding Base	E Rigid Sway Strut	SC Snubber		
F-A Supports	F1.10	Class 1 Piping Supports	Visual, VT-3		CS: 1 FW: 12 HPCI: 3 MS: 16 RC: 8 RCIC: 15 RHR: 4 RWC: 5 SLC: 4	CS: 3 FW: 3 MS: 6 RHR: 8 RWC: 2		FW: 3 RCIC: 2	CS: 6 FW: 8 HPCI: 3 MS: 24 RC: 22 RHR: 8 RWC: 4		10
	F1.20	Class 2 Piping Supports	Visual, VT-3	CRD: 9 CS: 1 HPCI: 7 RHR: 15	CRD: 9 CS: 20 HPCI: 23 RHR: 109	CS: 5 HPCI: 11 RHR: 32	RHR: 3	CRD: 15 CS: 2 HPCI: 7 RHR: 20	CS: 4 HPCI: 10 RHR: 76		10
	F1.30	Class 3 Piping Supports	Visual, VT-3	CRV: 8 FPC: 7 RBCLC: 11 RBVC: 5 RCIC: 1 RHR: 13 SWS: 6	CRV: 38 ESW: 1 FPC: 30 RBCLC: 41 RBVC: 23 RCIC: 7 RHR: 50 SWS: 93	FPC: 4 RBCLC: 3 RCIC: 2 RHR: 3 SWS: 1	RHR: 6 SWS: 1	CRV: 2 FPC: 1 RBCLC: 8 RBVC: 1 RCIC: 1 RHR: 8	RCIC: 2 RHR: 4		10

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components	Relief Request/ Technical Approach & Position Number	Notes
F-A Supports (Continued)	F1.40	Supports Other Than Piping Supports (Class 1, 2, 3, or MC)	Visual, VT-3	CRD: 8 CS: 2 HPCI: 2 RC: 6 RCIC: 2 RHR: 30 RPV: 5 SWS: 2 16-PC: 1 16-TS: 43		10

Examination Category (with Examination Category Description)	Item Number (or Augmented Number)	Description	Examination Requirements	Total Number of Components by System	Relief Request/ Technical Approach & Position Number	Notes
R-A Risk-Informed Piping Examinations	R1.11	Welds Subject to Thermal Fatigue	Reference Notes Code Case N-716-1	CS: 6 FW: 16 HPCI: 1 MS: 4 RHR: 7 RWC: 4		1, 2, 3, 4
	R1.11/14	Welds Subject to Thermal Fatigue and Crevice Corrosion Cracking	Reference Notes Code Case N-716-1	FW: 5		
	R1.13	Welds Subject to Erosion Cavitation	Reference Notes Code Case N-716-1	RHR: 4		1, 2, 3, 4, 6
	R1.14	Welds Subject to Crevice Corrosion Cracking	Reference Notes Code Case N-716-1	CS: 1 FW: 3 RCIC:4		
	R1.14/16	Welds Subject to Crevice Corrosion Cracking and Intergranular or Transgranular Stress Corrosion Cracking (IGSCC or TGSCC)	Reference Notes Code Case N-716-1	RC: 10		
	R1.16	Welds Subject to Intergranular or Transgranular Stress Corrosion Cracking (IGSCC or TGSCC)	Reference Notes Code Case N-716-1	CRD: 1 CS: 1 NBI: 10 RC: 87 RHR: 6		2
	R1.20	Welds not Subject to a Damage Mechanism	Reference Notes Code Case N-716-1	CS: 37 FW: 47 HPCI: 23 MS: 115 NBI: 24 RC: 20 RCIC: 42 RHR: 27 RWC: 27 SLC: 33		

Note #	Note Summary
1	For the Fifth ISI Interval, JAF's ISI Class 1 and Class 2 piping inspection program will be governed by risk-informed regulations. The RI-ISI Program methodology is described in ASME Code Case N-716-1. The RI-ISI Program scope has been implemented as an alternative to the 2007 Edition through the 2008 Addenda of the ASME Section XI examination program for ISI Class 1 B-F and B-J welds and ISI Class 2 C-F-1 and C-F-2 welds in accordance with Code Case N-716-1.
2	Per ASME Code Case N-716-1, welds within the plant that are assigned to IGSCC Categories B through G will continue to meet existing IGSCC schedules, while IGSCC Category A welds have been subsumed into the RI-ISI Program.
3	Examination requirements within the RI-ISI Program are determined by the various degradation mechanisms present at each individual piping weld. Reference ASME Code Case N-716-1 for specific examination method requirements.
4	The RI-ISI Program scope includes welds in the BER piping, also referred to as the HELB region, which includes several non-class welds that fall within the BER augmented inspection program. NOTE: JAF has not identified any BER piping welds. .
5	Examination may be performed with the connection assembled and bolting in place under tension, provided the connection is not disassembled during the interval. If the bolting connection is disassembled for any reason during the interval, the examination shall be performed with the connection disassembled. Examination is required only once per interval.
6	Socket welds of any size and branch pipe connection welds NPS2 and smaller require only a VT-2 visual examination. The VT-2 visual examination shall be conducted during a system pressure test or a pressure test specific to that weld, in accordance with Examination Category B-P, C-H, or D-B, as applicable.
7	JAF is implementing Code Case N-716-1 which classifies components as either High Safety Significant (HSS) or Low Safety Significant (LSS). The components under Examination Category C-A and C-B were classified as LSS and therefore, require no examination.
8	Reserved
9	Examination Category B-O (Pressure-Retaining Welds In Control Rod Housings), Item Number B14.10 (Welds in CRD Housing) - the scope of examination is for pressure retaining welds in 10% of the peripheral CRD Housings. A total of 56 of the 185 CRD Housings are classified as peripheral components. Each CRD has two welds and JAF has selected the welds on 6 CRD Housings to be examined during the interval (10% of 56).
10	Snubber visual examinations and functional testing are performed in accordance with the ASME OM Code, Subsection ISTD. For a detailed discussion of the JAF Snubber Program, reference Section 4.2 of this document. The snubber attachments are still examined in accordance with ASME Section XI under Category F-A, Item No. F1.10-SC Snubber, F1.20-SC Snubber, and F1.30-SC Snubber.
11	These inspections are credited by the JAF License Renewal, BWR Vessel Internals Program. As such, they shall be considered a regulatory commitment as part of 10 CFR 54.
12	Ultrasonic thickness measurements shall be performed from the exterior surface of the Torus Shell in accordance with JAF calculation JAF-CALC-16-00008, Rev. 0, JAF Torus Corrosion Allowance. These examinations are being performed in support of the Torus Preservation Program. These examinations will be performed on a periodic basis.
13	JAF performs ultrasonic examination (UT) of the four Feedwater Nozzle Inner Radii on Nozzles N4A–D once every 10 years in accordance with GE-NE-523-A71-0594-A, Revisions 1.
14	As allowed by ASME Code Case N-613-1, JAF will perform a volumetric examination using a reduced examination volume (A-B-C-D-E-F-G-H) of Figures 1, 2, and 3 of the Code Case in lieu of the previous examination volumes of ASME Section XI, Figures IWB-2500-7(a), (b), and (c).
15	As allowed by ASME Code Case N-648-1 and Code Case N-702, JAF may perform a visual examination with enhanced magnification in lieu of a volumetric examination in ASME Section XI.
16	Reserved

Note #	Note Summary
17	Reserved
18	Reserved
19	In accordance with Relief Request RR-19, RPV circumferential welds were requested to be permanently deferred for the remaining term of the initial operating license. Permanent relief was requested to eliminate the ASME Section XI required volumetric examinations for the affected JAF components VC-1-2, VC-2-3, VC-3-4 and VC-4-BH-1. RR-19 was withdrawn by JAF Letter JAFP-16-0048 dated March 16. 2016, due to intent to cease power operation. With change of Ownership and guarantee of continued operation, this Relief Request was resubmitted and approved as RR-19 via JAF Letter JAFP-16-0137 dated September 8, 2016. This relief is good through October 17, 2034.
20	JAF will utilize the alternative requirements of ASME Code Case N-747 to provide the reactor vessel head-to-flange weld to be inspected by surface examination once each ten-year inspection interval, using the current surface examination area shown in Figure IWB-2500-5. This alternative requirement may only be implemented after the weld has received at least one inservice volumetric examination, which may be performed as part of the preservice inspection, with no service-induced flaws having been identified. Hence, there have been no defects detected at JAF on this weld during pre-service or inservice examinations. It is therefore concluded that the concurrent volumetric and surface examination requirement may be eliminated for the reactor vessel head-to-flange weld, and that the outer surface examination discussed above will be performed.

8.0 RELIEF REQUESTS FROM ASME SECTION XI

This section contains relief requests written per 10 CFR 50.55a(z)(1) for situations where alternatives to ASME Section XI requirements provide an acceptable level of quality and safety; per 10 CFR 50.55a(z)(2) for situations where compliance with ASME Section XI requirements results in a hardship or an unusual difficulty without a compensating increase in the level of quality and safety; and per 10 CFR 50.55a(g)(5)(iii) for situations where ASME Section XI requirements are considered impractical.

The following USNRC guidance was utilized to determine the correct 10 CFR 50.55a paragraph citing for JAF relief requests. 10 CFR 50.55a(z)(1) and 10 CFR 50.55a(z)(2) provide alternatives to the requirements of ASME Section XI, while 10 CFR 50.55a(g)(5)(iii) recognizes situational impracticalities.

10 CFR 50.55a(z)(1): Cited in relief requests when alternatives to the ASME Section XI requirements which provide an acceptable level of quality and safety are proposed. Examples are relief requests which propose alternative NDE methods and/or examination frequency.

10 CFR 50.55a(z)(2): Cited in relief requests when compliance with the ASME Section XI requirements is deemed to be a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Examples of hardship and/or unusual difficulty include, but are not limited to, excessive radiation exposure, disassembly of components solely to provide access for examinations, and development of sophisticated tooling that would result in only minimal increases in examination coverage.

10 CFR 50.55a(g)(5)(iii): Cited in relief requests when conformance with ASME Section XI requirements is deemed impractical. Examples of impractical requirements are situations where the component would have to be redesigned, or replaced to enable the required inspection to be performed.

An index for JAF relief requests is included in Table 8.0-1. The "JAF-I5R-XX" relief requests are applicable to ISI, CISI, SPT, and PDI.

The following relief requests are subject to change throughout the inspection interval (e.g., USNRC approval, withdrawal). Changes to USNRC approved alternatives (other than withdrawal) require USNRC approval.

TABLE 8.0-1
RELIEF REQUEST INDEX

Relief Request	Revision Date[1]	Status[2]	(Program) Description/ Approval Summary[3]
RR-19	1/9/2017	TAC NO. MF6616 This relief is a resubmittal of RR-19 for the deferral of the RPV Circ. Welds for the period of extended operation	Reactor Pressure Vessel Circumferential Shell Weld Examinations

TABLE 8.0-1
RELIEF REQUEST INDEX

Relief Request	Revision Date[1]	Status[2]	(Program) Description/ Approval Summary[3]
JAF-I5R-02, Revision 1	12/11/2017	Granted – CAC NO. MG0116, EPID L-2017-LLR-0083, Accession No. ML18039A854. NOTE: If JAF takes exceptions to, or deviations from, the NRC staff-approved BWRVIP inspection guidelines, this will require JAF to revise and re-submit the relief request and will be required to receive NRC approval prior to implementing the revised inspection guidelines. (Section 5 of the relief request)	Examine ASME Section XI reactor internals in accordance with BWRVIP guidelines.
JAF-I5R-03	12/11/2017	Granted – CAC NO. MG0117, EPID L-2017-LLR-0084 and Accession No. ML18039A854.	ASME Boiler & Pressure Vessel Code, Section XI, ISI Ferritic piping butt welds requiring radiography during repair/replacement activities.
JAF-I5R-04	12/11/2017	Granted – CAC NO. MG0118, EPID L-2017-LLR-0085 and Accession No. ML18039A854.	Proposed alternative from the requirement to perform in-service ultrasonic examinations of Examination Category B-G-1, Item Number B6.40, Threads in Flange.

TABLE 8.0-1
RELIEF REQUEST INDEX

Relief Request	Revision Date[1]	Status[2]	(Program) Description/ Approval Summary[3]
JAF-I5R-05	3/2/2018	Granted - ML1239A010	Proposed Alternative to Use ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds"
RR for N-789-1		Granted – CAC NO. MF9692 and ML17289A075	Proposed Alternative to Utilize ASME Code Case N-789-1
RR for N-513-4		Granted – CAC NO. MF9641 and ML17219A248	Proposed Alternative to Utilize ASME Code Case N-513-4
RR for N-879	Withdrawn 7/26/2018	JAFP-18-0052, submitted on May 30, 2018	Proposed Alternative to Utilize ASME Code Case N-879
RR for N-878 & N-880		JAFP-18-0053, submitted on May 30, 2018	Proposed Alternative to Utilize ASME Code Cases N-878 and N-880

Note 1: The revision listed is the latest revision of the subject relief request. The date this revision became effective is the date of the approving SER, which is listed in the fourth column of the table. The date noted in the second column is the date of the ISI Program Plan revision when the relief request was incorporated into the document.

Note 2: This column represents the status of the latest revision. Relief Request Status Options: Authorized - Approved for use in an USNRC SER (See Note 1); Granted - Approved for use in an USNRC SER (See Note 1); Authorized Conditionally - Approved for use in an USNRC SER which imposes certain conditions; Denied - Use denied in an USNRC SER; Expired - Approval for relief has expired; Withdrawn - Relief has been withdrawn by JAF; Not Required - The USNRC has deemed the relief unnecessary in an SER or RAI; Cancelled - Relief has been cancelled by JAF prior to issue; Drafted - Drafted relief awaiting submittal and/or pending approval; Submitted - Relief has been submitted to the USNRC by the station and is awaiting approval.

Note 3: The USNRC grants relief requests pursuant to 10 CFR 50.55a(g)(6)(i) when Code requirements cannot be met and proposed alternatives do not meet the criteria of 10 CFR 50.55(z). The USNRC authorizes relief requests pursuant to 10 CFR 50.55a(z)(1) if the proposed alternatives would provide an acceptable level of quality and safety or under 10 CFR 50.55a(z)(2) if compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of safety.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELIEF REQUEST NO. 19
ENTERGY NUCLEAR FITZPATRICK, LLC AND
ENTERGY NUCLEAR OPERATIONS. INC.
JAMES A. FITZPATRICK NUCLEAR POWER
PLANT DOCKET NO. 50-333

1.0 INTRODUCTION

By application dated September 8, 2016 (Reference 1), as supplemented by letter dated November 9, 2016 (Reference 2), Entergy Nuclear FitzPatrick, LLC (the licensee) submitted a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the reactor pressure vessel (RPV) shell welds at the James A. FitzPatrick Nuclear Power Plant (JAFNPP), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, paragraph 50.55a(z)(1).

Specifically, the licensee proposes to permanently eliminate the volumetric examination requirements of Section XI of the ASME Code for RPV circumferential welds (ASME Code, Section XI, Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," Item No. B1.11), for the remainder of JAFNPP's fourth ISI interval and through the period of extended operation (PEO). Details of the licensee's proposed alternative are in Section 3.3 of this safety evaluation (SE). Section 50.55a(z)(1) of 10 CFR requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety. Section 2.0 of this SE provides a discussion of the attendant regulations and requirements. The proposed alternative is based on Electric Power Research Institute proprietary report TR-105697, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)" (Reference 3), and follows the implementation guidance in the NRC staff's July 28, 1998, SE of the report (Reference 4).

2.0 REGULATORY REQUIREMENTS AND GUIDANCE

2.1 Requirements of 10 CFR

The RPV shell welds at JAFNPP are ASME Code, Class 1 components, whose ISI requirements are performed in accordance with Section XI, "Rules for lnservice Inspection of Nuclear Power Plant Components," of the ASME Code and applicable edition and addenda, as required by 10 CFR 50.55a(g). Pursuant to 10 CFR 50.55a(g)(4), 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 the ASME Code, Section XI, to the extent practical, within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(a)(1)(ii), 12 months prior to the start of the 120-month interval, subject to the limitations and modifications in 10 CFR 50.55a(b)(2). The Code of record for JAFNPP for the fourth 10-year ISI interval is the 2003 Addenda to the 2001 Edition of the ASME Code, Section XI.

2.2 NRC Staff's Safety Evaluation of BWRVIP-05

The technical basis for the proposed alternative is BWRVIP-05, which calculates conservative conditional probabilities of failures for RPV welds. The basic principle for justifying the proposed alternative is to demonstrate that the conditional probabilities of failures of specific RPV welds are lower than the conservative values determined in BWRVIP-05. By letter dated February 22, 2000 (Reference 5), the NRC staff issued an SE that approved the proposed alternative for JAFNPP's initial licensing term. However, Section 3, "Conclusions," of the July 28, 1998, SE of BWRVIP-05 stated that since the failure frequency for the limiting circumferential weld could significantly increase through the PEO, the NRC staff will be requesting plants to perform plant-specific assessments that consider weld chemistry and neutron fluence at the end of the PEO. In addition, the July 28, 1998, SE stated that licensees may also request relief from the requirements of the ASME Code, Section XI, Examination Category B-A, Item No. 81 .11 applicable through the end of the PEO by demonstrating the following:

(1) At the expiration of the license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC staff's July 28, 1998, SE of BWRVIP-05.

(2) Licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC staff's July 28, 1998, SE of BWRVIP-05. Section 4, "Implementation," of the July 28, 1998, SE stated that if the axial weld examinations reveal an active mode of degradation, the examination of the circumferential welds shall be performed.

3.0 TECHNICAL EVALUATION

a. ASME Code Requirements

The specific examination requirement for RPV shell welds is volumetric examination of essentially 100 percent of the weld length of the volume defined in Figure IWB-2500-1, "Vessel Shell Circumferential Weld Joints," of the ASME Code, Section XI, as specified in Table IWB-2500-1, "Examination Categories," of the ASME Code, Section XI, Examination Category B-A, Item No. 81 .11.

b. ASME Code Components Affected

ASME Code Class: 1
Examination Category: B-A
Item Number: B1.11
Component: RPV Circumferential Welds
Component Numbers: VC-1-2, VC-2-3, VC-3-4, and VC-4-BH-1

3.3 Licensee's Proposed Alternative

Pursuant to 10 CFR 50.55a(z)(1), the licensee proposes the following 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. 81 .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.

3.4 Licensee's Basis

3.4.1 Satisfying the Conditional Probability of Failure for Welds Through the Period of Extended Operation.

The licensee performed a time-limited aging analysis (TLAA) of the RPV circumferential welds through the PEO in its 2006 license renewal application (LRA) (Reference 6) for JAFNPP. The NRC staff approved the LRA in 2008, as documented in NUREG-1905, "Safety Evaluation Report Related to the License Renewal of James A. FitzPatrick Nuclear Power Plant" (Reference 7). The NRC staff's evaluation of the TLAA of the RPV circumferential welds through the end of the PEO in enclosed in Section 4.2.5 of NUREG-1905, "Reactor Vessel Circumferential Weld Inspection Relief'. The licensee included information from Section 4.2.5.2 of NUREG-1905, "Staff Evaluation," in Section 5 of its submittal. Specifically, the licensee presented information from Table 4.2.5-1 of NUREG-1905, "Comparison of 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," in Table 1 of the submittal. Table 1 of the submittal compares conditional probabilities of failures for the RPV circumferential welds, computed for a bounding case of 64 effective full power years (EFPY), from the SE of BWRVIP-05, dated July 28, 1998, with those from NRC staff's and licensee's calculations for JAFNPP for 54 EFPY. JAFNPP's operational condition at the end of the PEO is represented by 54 EFPY. Note 2 of Table 1 of the submittal includes the acceptance criterion for the conditional probability of failure that the NRC staff approved: if JAFNPP's mean reference temperature (nil ductility transition) (RT NDT) or the limiting RPV circumferential weld (weld 1-240) is less than the mean RT NDT of the bounding case, then JAFNPP's conditional probability for failure for weld 1-240 is less than that of the bounding case. Table 1 shows that since JAFNPP's mean RT NDT of 81.1 degrees Fahrenheit (°F) for weld 1-240 is less than that of the bounding mean RT NDT of 128.5 °F, the conditional probability of failure for weld 1-240 is less than that of the bounding case. Note 2 in Table 1 of the submittal further states that plants that meet the above acceptance criteria may conclude that the conditional probability of failure for the limiting RPV circumferential weld is low enough to justify elimination of the required ASME Code volumetric examinations. The examination of the circumferential welds shall be performed if the axial weld examinations reveal an active mode of degradation. In the submittal, the licensee included its response to a previous request for additional information (RAI) regarding confirmation on whether previous volumetric examinations of the RPV axial welds showed any indication of cracking or any other age-related degradation. The licensee's response to RAI 4.2.5-2 from NUREG-1905 stated that no unacceptable inservice examination indications have been found on the RPV circumferential or axial welds. The NRC staff noted that the licensee referenced the NRC staff's evaluation (Reference 5) for the conditional probabilities of failures for the RPV circumferential welds applicable to JAFNPP's initial licensing term.

3.4.2 Lower Head Events

Section 6 of the licensee's submittal includes information on two recent lower head events that violated the JAFNPP Limiting Conditions for Operation (LCOs) in Section 3.4.9 of the Technical Specifications (TSs). Specifically, during any 1-hour period, the subject lower head events caused a violation of the TS requirement for reactor coolant system temperature change during heatup or cooldown of s 100 °F. The licensee stated that the two lower head events were entered in its correction action program, evaluated, and found to be acceptable. Evaluations of the two lower head events were also included as enclosures in the submittal.

3.5 NRC Staff's Evaluation

3.5.1 Satisfying the Conditional Probability of Failure for Welds Through the Period of Extended Operation.

Circumferential Welds As mentioned in Section 3.4.1 of this SE, the licensee included Table 4.2.5-1 of NUREG-1905 as Table 1 in its submittal. Table 1 contains the information to justify the elimination of the JAFNPP RPV circumferential welds during the PEO, based on BWRVIP-05. Furthermore, the licensee has shown that the JAFNPP conditional probability of failure for the limiting RPV circumferential weld is bounded by the BWRVIP-05 analysis and, therefore, elimination of the required ASME Code volumetric examinations through the PEO is justified. The NRC staff approved the licensee's evaluation in Table 1 in 2008. The NRC staff notes that surveillance capsule test data that are withdrawn and/or tested after 2008 can potentially impact the values in Table 1 and invalidate them, especially values of mean RT Nor. One of the conditions listed in Section 2.V of the JAFNPP Renewed License No. DPR-59, "Capsule withdrawal schedule," is that any changes to the capsule withdrawal schedule must be approved by the NRC prior to implementation. Furthermore, any changes to the capsule withdrawal schedule may also impact the values in Table 1. JAFNPP's capsule withdrawal schedule is contained in the integrated surveillance program (ISP) in BWRVIP-116 (Reference 8), as indicated in JAFNPP's updated final safety analysis report (UFSAR). Section 16.10.1.26, "Reactor Vessel Surveillance Program," indicates that JAFNPP's surveillance schedule includes the PEO. On October 18, 2016 (Reference 9), the NRC staff issued an RAI to confirm whether there have been any changes to the JAFNPP surveillance capsule withdrawal schedule or any surveillance test results since 2008 that could invalidate the technical basis for the proposed alternative. By letter dated November 9, 2016 (Reference 2), the licensee responded that there have been no changes to the capsule withdrawal schedule since the JAFNPP LRA in 2006 and clarified that JAFNPP's capsule withdrawal schedule is in accordance with Table 4-8, "ISP Test Matrix Results," in Revision 1-A of BWRVIP-86 (Reference 10). Revision 1-A of

BWRVIP-86 merged the information in BWRVIP-116 into a single updated ISP. The NRC staff verified that Table 4-8 in Revision 1-A of BWRVIP-86 is equivalent to Table 3-3 in BWRVIP-116. In addition, the licensee stated that it evaluated an ISP representative surveillance plate material in 201 O and documented the results in the ISP data source book, BWRVIP-135 (Reference 11). The licensee determined in its evaluation that there have been no changes to the JAFNPP RPV material properties nor to pressure-temperature limit curves. The NRC staff verified that testing of the 2010 ISP representative surveillance plate is in Table 4-7, "Detailed Test Plan by Plant," in Revision 1-A of BWRVIP-86 for JAFNPP.

The NRC staff determined that the values and evaluation in Table 1 of the licensee's submittal for the RPV circumferential welds are valid through the end of JAFNPP's PEO for the following reasons, as supported by the licensee's response to the NRC staff's RAI:

- The licensee has not changed its capsule withdrawal schedule since the JAFNPP LRA in 2006.
- The licensee has evaluated and documented surveillance capsule tests since 2008 and has determined that the tests do not impact the JAFNPP material properties, among which is mean RT NDT, the material property used directly in the criterion for acceptability of the conditional probability of failure values for the JAFNPP RPV circumferential welds. Therefore, the licensee has satisfied item 1 in the SE of BWRVIP-05.

Axial Welds According to Section 3.3 of this SE, the JAFNPP RPV axial welds will be examined in accordance with the requirements of Section XI of the ASME Code. With respect to conditional probabilities of failure, the licensee also performed a TLAA for the RPV axial welds through the PEO in its 2006 LRA. The NRC staff evaluated and approved it in Section 4.2.6 of NUREG-1905, "Reactor Vessel Axial Weld Failure Probability," in 2008 (Reference 7). The licensee has shown in the TLAA that the JAFNPP's conditional probability for failure for the axial welds through the PEO is acceptable. The NRC staff's review of the axial welds TLAA consisted of the validity of the analysis since the 2008 NRC staff approval. The NRC staff determined that the TLAA that evaluated the conditional probabilities of failure for RPV axial welds are still valid through JAFNPP's PEO for the same reasons the TLAA for the RPV circumferential welds are valid, as discussed above in Section 3.5.1 of this SE, under "Circumferential Welds." Regarding any signs of degradation in the RPV axial welds, the licensee referred to its response to RAI 4.2.5-2 in NUREG-1905, which stated that, "no unacceptable inservice examination indications have been found on reactor vessel welds (circumferential or axial)." The NRC staff has found the response acceptable. Therefore, the licensee has satisfied the implementation section of the SE of BWRVIP-05.

3.5.2 Lower Head Events

In the submittal, the licensee included two events in the RPV lower head that violated the JAFNPP TS requirements of s 100 °F over any 1-hour period during heatup and cooldown. In both events, instrumentation in the RPV bottom head recorded a maximum increase in metal temperature over a 1-hour period of 124.9 °F for one event and 125.7 °F for the other event. The NRC staff considers both events as heatup events because the temperature increased in both events. The NRC staff's evaluation of these two lower head events focuses on its potential impact on the proposed alternative to eliminate the ASME Code examination requirements of the RPV circumferential shell welds through the PEO. Heatup events generate compressive stresses on the inside surface of the RPV shell welds. A heatup event would, therefore, have no adverse impact on the values of conditional probability of failure for the type of flaws postulated in the BWRVIP-05 probabilistic fracture mechanics analysis. Therefore, the NRC staff determined that the two RPV lower head events included in the licensee's submittal have no relevant impact on the technical basis for the proposed elimination of the ASME Code volumetric examination requirements for the RPV circumferential shell welds at JAFNPP.

3.5.3 High Pressure Sources

For BWRVIP-05, item 2, the licensee identified the high pressure sources, which include the feedwater system, high pressure coolant injection (HPCI) system, reactor core isolation cooling (RCIC) system, control rod drive (CRD) system, reactor water cleanup (RWCU) system, and the standby liquid control (SLC) system. The NRC staff reviewed JAFNPP's UFSAR and determined that the licensee has identified the high pressure injection sources at the plant. The licensee stated that the feedwater system, HPCI system, and RCIC system are steam turbine driven and, therefore, it is not plausible for these systems to contribute to an over-pressurization event while the unit is in cold shutdown. The NRC staff has reviewed the JAFNPP's UFSAR and determined that these systems identified by the licensee are steam turbine driven and, therefore, it is not plausible that they will contribute to an over-pressurization event while the unit is in cold shutdown. The licensee stated that the SLC system has no automatic starts associated with the system and requires the operators to manually start the system from the control room or from the local test station. The licensee also stated that the injection rate of the SLC pump is approximately 50 gallons per minute, which gives the operators ample time to control reactor pressure in the case of an inadvertent injection. The NRC staff reviewed JAFNPP's UFSAR and determined that the SLC system, which is driven by two positive-displacement pumps, requires a manual operator action to initiate the system, and the injection rate provided by the licensee is consistent with the UFSAR. Given that the SLC is only operated in plant emergency situations or during testing conditions, and the injection rate is small relative to the total volume of the reactor vessel, the NRC staff determined that the licensee adequately justified that the design, training, and procedures, limit the cold over-pressure event with respect to the SLC system. The licensee stated that during normal cold shutdown conditions, RPV level and pressure are controlled with the CRD and RWCU systems using a "feed and bleed" process and the RPV is not taken water solid during these times. Additionally, "feed and bleed" is used during pressure testing of the RPV. The NRC staff reviewed the applicable sections of the UFSAR, TSs, and TS Basis and determined that the design, training, and procedures limit the cold over-pressure event with respect to the CRD and RCWU systems.

The NRC staff determined that the licensee identified all sources of high pressure injection and adequately justified that the procedures and training limit the cold over-pressure events. Additionally, the NRC staff compared this relief request to a previous relief request (Reference 5), and determined that there are no additional high pressure injection sources or changes to the procedures or training that would increase the likelihood of the cold over-pressure event. Based on its review, the NRC staff determined that the licensee satisfied item 2 in the SE of BWRVIP-05.

4.0 CONCLUSION

As set forth above, the NRC staff determined that the licensee's proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the staff authorizes the proposed alternative for the remainder of the fourth ISI interval, and through the PEO at JAFNPP, which ends on October 17, 2034. All other requirements of Section XI of the ASME Code for which relief was not specifically requested and approved in the subject relief requests remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

5.0 REFERENCES

1. Drews, W.C., Entergy Nuclear Operations, Inc., letter to the U.S. Nuclear Regulatory Commission, "Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1), Implementation of BWRVIP-05 at James A. FitzPatrick Nuclear Power Plant," dated September 8, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 16252A473).

2. Drews, W.C., Entergy Nuclear Operations, Inc., letter to the U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information - Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1), Implementation of BWRVIP-05 at James A. FitzPatrick Nuclear Power Plant," dated November 9, 2016 (ADAMS Accession No. ML 16314E532).

3. Electric Power Research Institute Report TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," September 1995 (EPRI Proprietary) (ADAMS Accession No. ML032200246 (non-proprietary version)).

4. Lainas, Gus C., U.S. Nuclear Regulatory Commission, letter to Carl Terry, BWRVIP Chairman, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925)," dated July 28, 1998 (ADAMS Legacy Library No. 9808040037).

5. Gamberoni, Marsha K., U.S. Nuclear Regulatory Commission, letter to James Knubel, Power Authority of the State of New York, "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)," dated February 22, 2000 (ADAMS Accession No. ML003685801).

6. James A. FitzPatrick Nuclear Power Plant License Renewal Application, dated July 31, 2006 (ADAMS Package Accession No. ML062140129).

7. U.S. Nuclear Regulatory Commission Report NUREG-1905, "Safety Evaluation Report Related to the License Renewal of James A FitzPatrick Nuclear Power Plant," April 2008 (ADAMS Accession No. ML081510826).

8. Electric Power Research Institute Report TR-1007824, "BWRVIP-116: BWR Vessel and Internals Project, Integrated Surveillance Program (ISP), Implementation for License Renewal," July 2003 (EPRI Proprietary).

9. Render, Diane, U.S. Nuclear Regulatory Commission, letter to the James A FitzPatrick Nuclear Power Plant Site Vice President, Entergy Nuclear Operations, Inc., "James A FitzPatrick Nuclear Power Plant- Request for Additional Information Re: Relief Request for Proposed Alternative for the Implementation of BWRVIP-05 (CAC No. MF8361),"dated October 18, 2016 (ADAMS Accession No. ML 16280A573).

10. Electric Power Research Institute Report TR-1025144, "BWRVIP-86, Revision 1-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP),

Implementation Plan," October 2012 (EPRI Proprietary).

11. Electric Power Research Institute Report TR-3002003144, "BWRVIP-135, Revision 3: BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations," December 2014. Principal Contributors: D. Dijamco J. Borromeo

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

May 30, 2018

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT-RELIEF REQUESTS I5R-02, I5R-03, AND I5R-04 FOR ALTERNATIVES TO CERTAIN ASME CODE REQUIREMENTS (CAC NOS. MG0116, MG0117, AND MG0118; EPID L-2017-LLR-0083, EPID L-2017-LLR-0084, AND EPID L-2017-LLR-0085)

Dear Mr. Hanson:

By letter dated August 10, 2017, as supplemented by letter dated December 11, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML 17223A280 and ML 17346A 153, respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted three relief requests to the U.S. Nuclear Regulatory Commission (NRC) for relief from certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), Section XI requirements at the James A. FitzPatrick Nuclear Power Plant. The relief requests are associated with the fifth inservice inspection interval, which began on August 1, 2017, and is currently scheduled to end on June 15, 2027.

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternatives on the basis that the alternatives provide an acceptable level of quality and safety.

The NRC staff has reviewed the subject requests and concludes, as set forth in the enclosed safety evaluations, that Exelon has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1).

All other ASME Code requirements for which relief was not specifically requested and approved in the subject requests remain applicable.

10 CFR 50.55a RELIEF REQUESTS: I5R-02, I5R-03, I5R-04

B. Hanson

If you have any questions, please contact Tanya Hood at 301-415-1387 or Tanya.Hood@nrc.gov.

Sincerely,

James G. Danna, Chief
Plant Licensing Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures:
1. Safety Evaluation for Relief Request I5R-02
2. Safety Evaluation for Relief Request I5R-03
3. Safety Evaluation for Relief Request I5R-04

cc: Listserv

10 CFR 50.55a RELIEF REQUESTS: I5R-02, I5R-03, I5R-04

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELIEF REQUEST I5R-02, REVISION 0, TO ALLOW USE OF BOILING WATER REACTOR VESSEL AND INTERNALS PROJECT GUIDELINES

EXELON GENERATION COMPANY, LLC JAMES A. FITZPATRICK NUCLEAR POWER PLANT DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated August 10, 2017, as supplemented by letter dated December 11, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML17223A280 and ML17346A 153, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted Relief Request I5R-02, Revision 0, to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for its fifth 10-year inservice inspection (ISI) interval regarding inspection of its reactor vessel internals (RVIs) components at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick). In this safety evaluation, the term "RVI components" includes reactor vessel (RV) interior surfaces, attachments, and core support structures.

Specifically, in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee proposed to use Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines as an alternative to certain requirements of Section XI, "Rules for ISI of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for ISI of RVI components. The regulation at 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety.

2.0 REGULATORY REQUIREMENTS

Section 50.55a(g) of 10 CFR states, in part, that ISI of certain ASME Code Class 1, 2, and 3 systems and components be performed in accordance with Section XI of the ASME Code and applicable addenda incorporated by reference in the regulations, as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained.

Section 50.55a(z) of 10 CFR states, in part, that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC prior to

10 CFR 50.55a RELIEF REQUESTS: I5R-02, I5R-03, I5R-04

implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative 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. Section 50.55a of 10 CFR allows the NRC to authorize alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

The regulations require that inservice examination of components and system pressure tests conducted during the successive 120-month inspection intervals (following the initial 120-month inspection interval) must comply with the requirements in the latest edition and addenda of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(a) 12 months before the start of the 120-month interval (or the optional ASME Code Cases listed in NRC Regulatory Guide (RG) 1.147, Revision 17, "lnservice Inspection Code Case Acceptability, ASME Section XI, Division 1"), subject to the conditions listed in 50.55a(b). The applicable ASME Code of record for the fifth 10-year ISI interval for FitzPatrick is ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

3.0 LICENSEE'S EVALUATION

3.1 Applicable Code Requirements

The ASME Code, Section XI, requires a visual examination (VT) of certain RVI components. These examinations are included in Table IWB-2500-1, Categories B-N-1 and B-N-2, and identified with the following item numbers:

- B13.10 - Examine accessible areas of the RV interior during each period using a VT-3 examination, as defined in paragraph IWA-2213 of Section XI of the ASME Code.

- B13.20 - Examine interior attachment welds within the RV beltline region during each interval using a VT-1 examination, as defined in paragraph IWA-2211 of Section XI of the ASME Code.

- B13.30 - Examine interior attachment welds outside of the beltline region during each interval using a VT-3 examination, as defined in paragraph IWA-2213 of Section XI of the ASME Code.

- B13.40 - Examine accessible surfaces of the welded core support structures during each interval using a VT-3 examination, as defined in paragraph IWA-2213 of Section XI of the ASME Code.

These examinations are performed to periodically assess the structural integrity of the RV interior surfaces, attachments, and core support structures.

3.2 Components for Which Relief is Requested

The ASME Code, Section XI, Class 1, Examination Categories B-N-1 and B-N-2, Code Item Numbers B13.10, Vessel Interior; B13.20, Interior Attachments within Beltline Region; B13.30, Interior Attachments Beyond Beltline Region; and B13.40, Core Support Structure.

3.3 Licensee's Reason for Request

10 CFR 50.55a RELIEF REQUESTS: I5R-02, I5R-03, I5R-04

The licensee stated that implementation of the alternative inspection program will maintain an adequate level of quality and safety of the affected welds and components and will not adversely impact the health and safety of the public. As part of its justification for the relief, the licensee stated that boiling-water reactors (BWRs) now examine the RV interior surfaces, attachments, and core support structures in accordance with BWRVIP inspection and evaluation (I&E) guidelines in lieu of ASME Code, Section XI criteria. The BWRVIP guidelines are written for the safety-significant RVI components and provide appropriate examination and evaluation criteria using appropriate methods and reexamination frequencies. The proposed alternative includes examination methods, examination volume, frequency, training, successive and additional examinations, flaw evaluations, and reporting. Furthermore, the licensee stated that this alternative to the ASME Code, Section XI requirements is requested pursuant to 10 CFR 50.55a(z)(1).

3.4 Licensee's Proposed Alternative and Basis for Use

In lieu of the requirements specified in Section XI of the ASME Code, the licensee proposed to examine the FitzPatrick RVI components in accordance with BWRVIP I&E guideline requirements in the following BWRVIP reports for RV surfaces, attachments, and core support structures.

- BWRVIP-03, "BWRVIP Reactor Pressure Vessel and Internals Examination Guidelines"
- BWRVIP-18, Revision 1-A, "BWRVIP Core Spray Internals Inspection and Flaw Evaluation Guidelines"
- BWRVIP-25, "BWRVIP Core Plate Inspection and Flaw Evaluation Guidelines"
- BWRVIP-26-A, "BWRVIP Top Guide Inspection and Flaw Evaluation Guidelines"
- BWRVIP-27-A, "BWRVIP BWR Standby Liquid Control System/Core Plate Delta P Inspection and Flaw Evaluation Guidelines"
- BWRVIP-38, "BWRVIP Shroud Support Inspection and Flaw Evaluation Guidelines"
- BWRVIP-41, Revision 3, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines"
- BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
- BWRVIP-48-A, "Vessel ID [Inside Diameter] Attachment Weld Inspection and Flaw Evaluation Guidelines"
- BWRVIP-76, Revision 1-A, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines"
- BWRVIP-138, Revision 1-A, "Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines"
- BWRVIP-180, "Access Hole Cover Inspection and Flaw Evaluation Guidelines"
- BWRVIP-183, "Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines"

10 CFR 50.55a RELIEF REQUESTS: I5R-02, I5R-03, I5R-04

- BWRVIP-94, Revision 2, "BWRVIP Program Implementation Guide"

The licensee stated that inspection services by an authorized inspection agency will be applied to the proposed alternative actions of this relief request. The licensee further indicated that results of examinations and deviations for the BWR fleet are reported under an established protocol between the BWRVIP and the NRC. Also, since the BWRVIP guidelines are revised periodically, the licensee clarified that if new guidance includes changes that are less conservative than those approved by the NRC, this less conservative guidance shall be implemented only after NRC approval.

The licensee provided a comparison of the ASME Code, Section XI examination requirements for Categories B-N-1 and B-N-2 for RV surfaces, attachments, and core support structures with the current BWRVIP guideline requirements, as applicable to FitzPatrick in Table 1 of its submittal. In Enclosure 1 of its submittal, the licensee provided additional justification regarding the comparison of the inspection requirements of the ASME Code, Section XI, Table IWB-2500-1, Item Numbers B13.10, B13.20, B13.30, and B13.40, to the inspection requirements in the BWRVIP guidance documents. For example, the following excerpt from Enclosure 1 of its submittal indicates the applicable ASME Code, Section XI category/item numbers that are applicable to some of the FitzPatrick RVI components:

- Core Spray Piping, Top Guide, Jet Pump Welds and Components, etc. - Item No. B13.10

- Jet Pump Riser Brace-to-RV Wall Pad Welds - Item No. B 13.20

- Core Spray Piping Bracket Welds - Item No. B13.30

- Core Shroud - Item No. B13.40

Based on examination method, scope, frequency, and flaw evaluation criteria, the licensee stated that the above examples demonstrate that the inspection techniques recommended by the BWRVIP I&E guidelines are equivalent to, or superior to, the inspection techniques mandated by the ASME Code, Section XI ISI program. For instance, the BWRVIP's inspection of jet pump riser braces per BWRVIP-41 uses enhanced VT-1 (EVT-1), whereas the ASME Code uses VT-1. The BWRVIP's inspection of core spray piping bracket welds per BWRVIP-48-A uses EVT-1 every 8 years for plants with a 2-year fuel cycle, whereas the ASME Code uses VT-3 every 10 years.

4.0 NRC STAFF EVALUATION

The NRC staff reviewed the information in Relief Request I5R-02, Revision 0, including the supplemental information in the letter dated December 11, 2017. The NRC staff reviewed the status of the referenced BWRVIP reports and found application of the referenced BWRVIP reports to be acceptable, provided that the NRC conditions associated with the latest safety evaluation for each BWRVIP report are implemented. By e-mail dated November 29, 2017 (ADAMS Accession No. ML 17335A 100), the NRC staff issued a request for additional information (RAI) requesting the licensee to explain why the current basis for examining the core plate would not be sufficient to manage either stress relaxation or cracking of the core plate rim hold-down bolts during the period of extended operation. The RAI request also asked for clarification regarding inspection results for Item B13.10 and justification for why BWRVIP-139 is not needed for either steam dryer hold-down brackets or steam dryer support brackets. Following is the NRC staff's evaluation of the licensee's response.

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4.1 Comparison of ASME Examination Category B-N-1 Requirements with BWRVIP Guidance Requirements

Except for the B13.10 component (RV interior), which belongs to Examination Category B-N-1, all other subject components in Table 1 of the licensee's submittal belong to Examination Category B-N-2. For the Category B-N-1 RV interior, it should be noted that portions of the various examinations required by the applicable BWRVIP guidelines require access to accessible areas of the RV during each refueling outage. Examination of core spray piping and spargers (BWRVIP-18, Revision 2-A), top guide (BWRVIP-26-A), jet pump welds and components (BWRVIP-41, Revision 3), interior attachments (BWRVIP-48-A), core shroud welds (BWRVIP-76, Revision 1-A), shroud support (BWRVIP-38), and lower plenum components (BWRVIP-47-A), all provide such access. Examining specific welds and components within the RV interior above and below the core and the surrounding annulus area using remote camera essentially performs equivalent VT-3 examination of many areas in the RV interior. This visual examination per BWRVIP reports is more frequent than that required by ASME Code, Section XI. The licensee further stated that evidence of wear structural degradation; loose, missing, or displaced parts; foreign materials; and corrosion product buildup have been observed during the course of implementing these BWRVIP examination requirements.

Based on its review, the NRC staff finds that the specified BWRVIP guideline requirements meet the subject Code requirements for examination method and frequency of the RV interior. Therefore, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety.

4.2 Comparison of ASME Examination Category B-N-2 Requirements with BWRVIP Guidance Requirements

Regarding the Table 1 comparison of the current ASME Code, Section XI examination requirements with the current BWRVIP guideline requirements for Category B-N-2 items of the licensee's submittal, the NRC staff noted that for Item B13.20, the proposed BWRVIP examination methods are EVT-1 for one component and VT-1 for another, as opposed to VT-1 specified in the ASME Code, Section XI, for both components. Similarly, for Items B13.30 and B 13.40, the proposed BWRVIP examination methods for the majority of the components are EVT-1 or ultrasonic testing, as opposed to VT-3 specified in the ASME Code, Section XI. For the examination frequency, Table 1 in the submittal shows that except for one Item B13.20 component, which has a longer examination interval (12 years versus 10 years), all other B13.20, B13.30, and B13.40 components have equivalent or shorter examination intervals. For this single B13.20 component, the slightly longer examination interval is compensated for by the better EVT-1 examination method. Therefore, for both the examination methods and frequency, the BWRVIP guidelines are equivalent or exceed the ASME Code, Section XI requirements.
Table 1information in the submittal regarding integrally welded core support structure under Item B13.40 requires further evaluation because BWRVIP-25 is not listed as "Applicable BWRVIP Document," and the core plate is not in the "BWRVIP Exam Scope." It should be noted that two approaches in managing the core plate integrity in BWRVIP-25 have been accepted in Section 3.0.3.2. 7, "BWR Vessel Internal Programs," of the license renewal safety evaluation for FitzPatrick, "Safety Evaluation Report Related to the License Renewal of James A. FitzPatrick Nuclear Power Plant" (ADAMS Accession No. ML080250372). They are (1) to install core plate wedges prior to the period of extended operation and (2) to complete a plant-specific analysis to determine acceptance criteria for continued inspection of the core plate rim hold-down bolting in accordance with BWRVIP-25. In the RAI e-mail dated November 29, 2017, the NRC staff asked the licensee in RAI-1 to confirm whether it needs to revise Table 1 by including BWRVIP-25 as one of the applicable BWRVIP documents for Item B13.40 so that it can perform either BWRVIP-25 option approved in the license renewal safety evaluation for FitzPatrick to manage the core

plate integrity. In its RAI response dated December 11, 2017, the licensee added BWRVIP-25 to Table 1 and provided Relief Request I5R-02, Revision 1.

Based on its review, the NRC staff finds that the proposed use of BWRVIP-25 guidance is another example of exceeding the ASME Code, Section XI, Table IWB 2500-1, B-N-2 requirements. Therefore, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety.

4.3 Operating Experience and Flaw Evaluation

4.3.1 Reactor Internals Inspection History

The NRC staff reviewed the information in Relief Request I5R-02, Revision 0, including the supplemental information in the December 11, 2017, submittal. The NRC staff reviewed the reactor internals inspection history to assess the impact of using the BWRVIP inspections and disposition of indications on FitzPatrick RVI integrity. The NRC staff found no inspection record for ASME Code Item B13.10, "Reactor Vessel Interior." In its RAI e-mail dated November 29, 2017, the NRC staff asked the licensee in RAl-2 to clarify whether the absence of inspection results for ASME Code Item B13.10 meant that no relevant indications were noted for this item in all past examinations. In its RAI response dated December 11, 2017, the licensee confirmed that a review of examination results since 2000 for Item B13.10 has not identified any indications that were rejected by the ASME Code, Section XI.

Based on its review, the NRC staff finds this response acceptable. Therefore, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety.

4.3.2 Flaw Evaluation Guidelines and Plant-Specific Leakage Assessment

The licensee does not mention the evaluation criteria for B13.20 components (B-N-2) and for the B13.40 component (B-N-2) core shroud. Enclosure 1 to Relief Request I5R-02, Revision 0, states, without elaboration, that comparable flaw evaluation criteria were used. The NRC staff examined the part of BWRVIP-48-A relevant to the B13.20 components, and the part of BWRVIP-38 and BWRVIP-76, Revision 1-A, relevant to the B13.40 components, and confirmed that the evaluation criteria for them, although not identical to the ASME Code, Section XI, were accepted by the NRC staff based on technical equivalency. It should be noted that although BWRVIP-38 does not have an "A" affix, a final safety evaluation for it was issued on July 24, 2000 (ADAMS Accession No. ML003735498). Further, Enclosure 2 to Relief Request I5R-02 indicates that historically, all indications in various RVI components were satisfactorily dispositioned by repair or evaluations, and all follow-up inspections showed no meaningful change. To further address cracking in several B-N-1 and B-N-2 components at FitzPatrick, as shown in Enclosure 2 to Relief Request I5R-02, Revision 0, the licensee performed plant-specific leakage assessments in accordance with BWRVIP requirements for identified or postulated through-wall cracking in (1) the core spray 190-degree downcomer weld, (2) jet pump diffuser welds, and (3) core shroud welds. Based on conservative assumptions, leakages were calculated to be 40 gallons per minute (gpm) through the core spray 190-degree downcomer crack repair, 98.5 gpm for the jet pump welds, and 205 gpm for all known and postulated core shroud cracking. They are within the allowable limits of 123 gpm for Case (1) and 200 gpm for Case (2). For Case (3), the licensee used shroud leakage as a direct input to the FitzPatrick loss-of-coolant accident analysis, and the results indicated that there is no increase to the peak cladding temperature. The licensee stated that this plant-specific peak cladding temperature is below the 10 CFR 50.46(b) regulatory limit of 2,200 degrees Fahrenheit (°F).

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Flaw evaluations are not required for B-N-1 components because the purpose of the examination is not to detect flaws, but rather to identify conditions such as distortion or displacement of parts; loose, missing, or fractured fasteners; foreign material; corrosion; erosion; wear; and structural degradation. The flaw evaluation methodologies for various B-N-2 components in the referenced BWRVIP reports are either the ASME Code, Section XI methodologies or accepted by the NRC staff based on acceptable levels of quality and safety. Subsequent inspections of the RVI components at FitzPatrick using the relevant BWRVIP I&E guidelines will provide reasonable assurance that emerging aging effects will be identified in a timely manner because (1) the FitzPatrick RVI inspection program has been developed and implemented to meet the requirements of the relevant BWRVIP reports, and (2) the BWRVIP I&E guidelines require the same or more frequent inspections than ASME Code, Section XI criteria for RVI components that are susceptible to aging degradation mechanisms.

In addition, frequent inspections in accordance with the BWRVIP I&E guidelines will enable the licensee to effectively monitor existing aging degradation in RV surfaces, attachments, and core support structures during the fifth ISI interval. For the associated plant-specific leakage assessments, the NRC staff concludes that they are acceptable because the leakage through the conservatively postulated core shroud cracks, combined with leakage in jet pump welds and the core spray weld, would not increase the peak cladding temperature analyzed in the FitzPatrick loss-of-coolant accident analysis.

For B13.30 components (B-N-2), Enclosure 1 to Relief Request I5R-02, Revision 0, indicated that for the interior attachment welds that require VT-3 examination, the ASME Code, Section XI flaw evaluation criteria is employed (BWRVIP-48-A). This is also true for core spray piping bracket welds.

4.4 Additional Technical Findings

In addition to the above evaluations, the NRC staff also has the following findings:

- Although furnace-sensitized stainless steel vessel attachment welds tend to be more susceptible to intergranular stress-corrosion cracking, the NRC staff's approval of the BWRVIP-48-A I&E guidelines is an indication that the alternative monitoring of intergranular stress-corrosion cracking in this type of welds is acceptable.

- The licensee did not include BWRVIP-139, "BWR Vessel Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines," to monitor active aging degradation in the steam dryer. In its RAI e-mail dated November 29, 2017, the NRC staff asked the licensee in RAI-3 to justify why BWRVIP-139 is not needed for either steam dry hold-down brackets or steam dryer support brackets listed in Table 1 of its submittal under Item B13.30. In its RAI response dated December 11, 2017, the licensee explained that the guidance for the steam dryer hold-down or support brackets is contained in BWRVIP-48. Therefore, BWRVIP-139 is not needed for steam dryer support brackets listed in Table 1 under Item B13.30.

- The licensee did not include BWRVIP-42, Revision 1, "BWR Vessel Internals Project, Low Pressure Coolant Injection System (LPCI) Coupling Inspection and Flaw Evaluation Guidelines," to monitor active aging degradation in LPCI couplings. This is acceptable because BWRVIP-42, Revision 1, does not apply to older BWR/4 plants such as FitzPatrick.

- In addition to BWRVIP-25, some BWRVIP reports that are included in this relief request do not have approved "A" versions either. This is appropriate because use of the specific I&E guidelines in these BWRVIP reports for the ASME Code, Section XI, Examination Categories B-N-1 and B-

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N-2, Code Item Numbers B13.10 to B13.40 RVI components have already been accepted by the NRC staff in prior applications, as indicated in the June 30, 2014, safety evaluation for Grand Gulf Nuclear Station, Unit 1 (ADAMS Accession No. ML 14148A262), and the March 10, 2016, safety evaluation for Clinton Power Station, Unit 1 (ADAMS Accession No. ML 16012A344).

These findings clarified the scope of the applicable BWRVIP reports.

5.0 CONCLUSION

As set forth above, the NRC staff determines that for Relief Request I5R-02, Revision 0, the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for Relief Request I5R-02, Revision 0. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year ISI interval, which began on August 1, 2017, and is currently scheduled to end on June 15, 2027.

The NRC staff notes that if the licensee intends to take exceptions to, or deviations from, the NRC staff-approved BWRVIP inspection guidelines, this will require the licensee to revise and re-submit this relief request. The licensee shall obtain NRC staff approval for such exceptions prior to implementing the revised inspection guidelines for the FitzPatrick unit's RV interior surfaces, attachments, and core support structures.

All other requirements of the ASME Code, Section XI, for which an alternative has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear lnservice Inspector. Any ASME Code, Section XI RVI components that are not included in this relief request will continue to be inspected in accordance with the ASME Code, Section XI requirements

Principal Contributor: Simon Sheng

Date: May 30, 2018

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WASHINGTON, D.C. 20555-
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELIEF

REQUEST I5R-03, REVISION 0, TO USE ENCODED PHASED ARRAY ULTRASONIC

EXAMINATION TECHNIQUES IN LIEU OF RADIOGRAPHY EXAMINATION

EXELON GENERATION COMPANY, LLC JAMES A.

FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated August 10, 2017, as supplemented by letter dated December 11, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML 17223A280 and ML 17346A153, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted Relief Request I5R-03, Revision 0, to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for its fifth 10-year inservice inspection (ISI) interval regarding the ferritic piping butt welds requiring radiography during repair/replacement activities at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick).

Specifically, in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee proposed to use an alternative that would allow the use of encoded phased array ultrasonic examination techniques (PAUT) in lieu of radiography (RT) examinations of ISI Class 1 and 2 ferritic piping repair/replacement welds required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for ISI of Nuclear Power Plant Components," at FitzPatrick. The regulation at 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Section 50.55a(g) of 10 CFR states, in part, that ISI of certain ASME Code Class 1, 2, and 3 systems and components be performed in accordance with Section XI of the ASME Code and applicable addenda incorporated by reference in the regulations, as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained.

Section 50.55a(z) of 10 CFR states, in part, that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC prior to implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative 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. Section 50.55a of 10 CFR allows the NRC to authorize alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

The regulations require that inservice examination of components and system pressure tests conducted during the successive 120-month inspection intervals (following the initial 120-month inspection interval) must comply with the requirements in the latest edition and addenda of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(a) 12 months before the start of the 120-month interval (or the optional ASME Code Cases listed in NRC Regulatory Guide (RG) 1.147, Revision 17, "lnservice Inspection Code Case Acceptability, ASME Section XI, Division 1"), subject to the conditions listed in 50.55a(b). The applicable ASME Code of record for the fifth 10-year ISI interval for FitzPatrick is ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

The guidance that the NRC staff considered in its review is NUREG/CR-7204, "Applying Ultrasonic Testing in Lieu of Radiography for Volumetric Examination of Carbon Steel Piping," September 2015 (ADAMS Accession No. ML 15253A674), which provides an initial technical evaluation of the capabilities of phased-array ultrasonic testing to supplant traditional radiographic testing for detection and characterization of welding fabrication flaws in carbon steel welds.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

3.0 LICENSEE'S EVALUATION

3.1 Applicable Code Requirements

The regulation in 10 CFR 50.55a(b)(2)(xx)(B) states that "[t]he NOE [nondestructive examination] provision in IWA-4540(a)(2) of the 2002 Addenda of Section XI must be applied when performing system leakage tests after repair and replacement activities performed by welding or brazing on a pressure retaining boundary using the 2003 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section."

- Subarticle IWA-4540(a)(2) of the 2002 Addenda of the ASME Code, Section XI, states, in part, that "the nondestructive examination method and acceptance criteria of the 1992 Edition or later of Section III be met prior to return to service." Subarticle IWA-4540(a)(2) must be completed in order to perform a system leakage test in lieu of a system hydrostatic test. The examination requirements for ASME Section III circumferential butt welds are contained in the ASME Code, Section III, Subarticles NB-5200, NC-5200, and ND-5200. The acceptance standards for radiographic examination are specified in Subarticles NB-5300, NC-5300, and ND-5300.

- Subarticle IWA-4221 requires that items used for repair/replacement activities meet the applicable Owner's Requirements and Construction Code requirements when performing repair/replacement activities.
- Subarticle IWA-4520 requires that welded joints made for installation of items be examined in accordance with the Construction Code identified in the Repair/Replacement Plan.

3.2 Components for Which Relief is Requested

The ASME Code, Section XI, requires RT of ferritic piping butt welds during repair/replacement activities.

3.3 Licensee's Reason for Request

The licensee stated that implementation of the use of encoded PAUT in lieu of RT to perform the required examinations of the replaced welds would eliminate the safety risk associated with performing RT, which includes the planned exposure and the potential for accidental personnel exposure. The proposed alternative minimizes the impact on other outage activities normally involved with performing RT such as limited access to work locations and the need to control system fill status because RT would require a line to remain fluid empty in order to obtain adequate examination sensitivity and resolution. As part of its justification for the relief, the licensee also stated that encoded PAUT has been demonstrated to be adequate for detecting and sizing critical flaws and replacement of piping is periodically performed in support of the flow-accelerated corrosion (FAC) program as well as other repair and replacement activities.

3.4 Licensee's Proposed Alternative and Basis for Use

The proposed alternative includes a qualification program that the NRC staff determined is substantially similar to ASME Code Case N-831, "Ultrasonic Examination in Lieu of Radiography for Welds in Ferritic Pipe," approved by the ASME Section XI Standards Committee on October 20, 2016. The differences between the proposed alternative and ASME Code Case N-831 were limited to editorial changes that clarified the wording.

The encoded PAUT procedures, equipment, and personnel will be qualified using performance demonstration testing. The flaw acceptance standards for the PAUT examinations will consider all flaws to be planar and they are evaluated against the preservice acceptance standards of Subarticles IWB-3400, IWC-3400, and IWD-3400 of the ASME Code, Section XI, for ASME Code Class 1, 2, and 3 welds, respectively.

The licensee stated that the basis for the proposed alternative is that encoded PAUT is equivalent or superior to RT for detecting and sizing planar flaws. The examination procedure and personnel performing examinations are qualified by performance demonstration testing using representative piping conditions and flaws that demonstrate the ability to detect and size flaws that are both acceptable and unacceptable to the defined acceptance standards. The licensee also states that ultrasonic testing (UT) techniques are being used throughout the nuclear industry for examination of dissimilar metal welds and overlaid welds, as well

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as other applications, including piping replacements covered under ASME B31.1, "Power Piping, ASME Code for Pressure Piping, B31."

4.0 NRC STAFF EVALUATION

The NRC staff reviewed the information in Relief Request I5R-03, Revision 0. The NRC staff assessed the effectiveness of the use of UT in lieu of RT since 2009 through literature reviews, detailed evaluations of previous relief requests and proposed alternatives, and confirmatory experimental work to validate findings. Ultrasonic and radiography testing are volumetric inspection techniques that are commonly used to inspect welds in nuclear power plants and in other industries. Ultrasonic testing examinations differ from RT examinations as they use different physical mechanisms to detect and characterize discontinuities. These differences in physical mechanisms result in several key differences in sensitivity and discrimination capability. An assessment of the use of UT in lieu of RT is described in NUREG/CR-7204. This report included evaluation on the use of UT in lieu of RT for welded pipes and plates with thicknesses ranging from 0.844 inches to 2.2 inches.

Based on its review, the NRC staff finds that there is a sufficient technical basis for the use of UT in lieu of RT for ferritic steel welds. Given that UT can be effective, the NRC staff considered whether the proposed alternative applies UT in a way that provides reasonable assurance of finding structurally-significant flaws.

Important aspects of the licensee's proposed alternative include:

- The examination volume shall include 100 percent of the weld volume and the weld-to-base-metal interface.

- The electronic data files for the PAUT examinations will be stored as archival-quality records. In addition, hard copy prints of the data will also be included as part of the PAUT examination records to allow viewing without the use of hardware or software.

- Ultrasonic testing examination procedures shall be qualified by using either a blind or a non-blind performance demonstration using a minimum of 30 flaws covering a range of sizes, positions, orientations, and types of fabrication flaws. The demonstration set shall include specimens to represent the minimum and maximum diameter and thickness covered by the procedure.

- The flaw through-wall heights for the performance demonstration testing shall be based on the applicable acceptance standards for volumetric examination in accordance with Subarticles IWB-3400, IWC-3400, or IWD-3400 of the ASME Code, Section XI. At least 30 percent of the flaws shall be classified as acceptable planar flaws, with the smallest flaws being at least 50 percent of the maximum allowable size based on the applicable aspect ratio for the flaw.

- Ultrasonic testing examination personnel shall demonstrate their capability to detect and size flaws by performance demonstration using the qualified procedure. The demonstration specimen set shall contain at least 10 flaws covering a range of sizes, positions, orientations, and types of fabrication flaws.

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- All flaws detected using angle-beam UT inspections will be treated as planar flaws and will be evaluated against the preservice acceptance standards in Subarticles IWB-3400, IWC-3400, and IWD-3400 of the ASME Code, Section XI, for ASME Code Class 1, 2, and 3 welds, respectively.

The NRC staff has authorized similar alternatives for other licensees, which include aspects similar to those listed above. The NRC staff finds that the use of performance demonstration for personnel and procedure qualification and the use of encoded data provide assurance that the PAUT methods will be sufficiently rigorous to detect and size flaws in the welds.

Currently, the licensee is required to use the RT acceptance standards in Section III of the ASME Code. Section III also provides UT acceptance standards; however, the licensee has requested to use the flaw acceptance standards in Section XI of the ASME Code as an alternative. The Section III RT and UT acceptance standards (Subarticles NB-5300, NC-5300, and ND-5300) require the inspector to detect and determine the type of flaw (e.g., porosity, lack of fusion, slag, incomplete penetration). While RT is effective at discerning between different flaw types, it is less capable than UT at detecting planar flaws such as cracks and lack-of-fusion defects. While Subarticles IWB-3400, IWC-3400, and IWD-3400 of Section XI of the ASME Code allow larger flaws than paragraphs NB-5330, NC-5330, and ND-5330 of Section III, the use of Section XI acceptance standards has proven effective for ISI of piping welds. The NRC staff finds that the use of the ASME Code, Section XI acceptance standards is appropriate for the proposed alternative, as the alternative is for repair/replacement activities, not new plant construction, and industry experience with Section XI acceptance standards has demonstrated their effectiveness.

Based on its review, the NRC staff finds that the use of the ASME Code, Section XI acceptance standards is appropriate for the proposed alternative, as the alternative is for repair/replacement activities, not new plant construction, and industry experience with Section XI acceptance standards has demonstrated their effectiveness. Therefore, the NRC staff concludes that the use of encoded PAUT qualified as proposed by the licensee for ferritic piping repair/replacement welds provides an acceptable level of quality and safety.

5.0 CONCLUSION

As set forth above, the NRC staff determines that for Relief Request I5R-03, Revision 0, the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for Relief Request I5R-03, Revision 0. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year ISI interval, which began on August 1, 2017, and is currently scheduled to end on June 15, 2027.

All other requirements of the ASME Code, Section XI, for which an alternative has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear lnservice Inspector. Any ASME Code, Section XI, RVI components that are not included in this relief request will continue to be inspected in accordance with the ASME Code, Section XI requirements.

Principal Contributor: Diane Render

Date: May 30, 201 8

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20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELIEF

REQUEST I5R-04, REVISION 0, TO PERFORM INSERVICE ULTRASONIC EXAMINATIONS OF

REACTOR PRESSURE VESSEL THREADS IN FLANGE EXAMINATION EXELON

GENERATION COMPANY, LLC

JAMES A. FITZPATRICK NUCLEAR POWER

PLANT DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated August 10, 2017, as supplemented by letter dated December 11, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML 17223A280 and ML 17346A153, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted Relief Request I5R-04, Revision 0, to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for its fifth 10-year inservice inspection (ISI) interval regarding Examination Category B-G-1, Item Number B6.40 threads in flange locations at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick).

Specifically, in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee proposed to use an alternative to certain requirements of Section XI, "Rules for ISI of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) to perform in-service ultrasonic examinations of Examination Category B-G-1, Item Number B6.40, Threads in Flange for the fifth 10-year ISI interval at FitzPatrick. The regulations in 10 CFR 50.55a(z)(1) require the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Section 50.55a(g) of 10 CFR states, in part, that ISI of certain ASME Code Class 1, 2, and 3 systems and components be performed in accordance with Section XI of the ASME Code and applicable addenda

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incorporated by reference in the regulations, as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained.

Section 50.55a(z) of 10 CFR states, in part, that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC prior to implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative 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. Section 50.55a of 10 CFR allows the NRC to authorize alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

The regulations require that inservice examination of components and system pressure tests conducted during the successive 120-month inspection intervals (following the initial 120-month inspection interval) must comply with the requirements in the latest edition and addenda of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(a) 12 months before the start of the 120-month interval (or the optional ASME Code Cases listed in NRC Regulatory Guide (RG) 1.147, Revision 17, "lnservice Inspection Code Case Acceptability, ASME Section XI, Division 1"), subject to the conditions listed in 50.55a(b). The applicable ASME Code of record for the fifth 10-year ISI interval for FitzPatrick is ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

3.0 LICENSEE'S EVALUATION

3.1 Applicable Code Requirements

The ASME Code, Section XI, requires a volumetric examination technique with 100 percent of the flange threaded stud holes examined every ISI interval for reactor pressure vessel (RPV) threads in flange, Examination Category B-G-1, Item Number B6.40. The examination area is defined in Figure IWB-2500-12, "Closure Stud and Threads in Flange Stud Hole," of the ASME Code, Section XI.

3.2 Components for Which Relief is Requested

The ASME Code, Section XI, Class 1, Examination Category B-G-1, Item Number B6.40 threads in RPV flange locations at FitzPatrick.

3.3 Licensee's Reason for Request

The licensee stated that an evaluation of potential degradation mechanisms that could impact flange/threads reliability was performed. Potential types of degradation evaluated included pitting, intergranular attack, corrosion fatigue, stress-corrosion cracking, crevice corrosion, velocity phenomena, dealloying corrosion, general corrosion, stress relaxation, creep, mechanical wear and mechanical/thermal fatigue. Other than the potential for mechanical/thermal fatigue, there are no active degradation mechanisms identified for the threads in flange component. The licensee described maintenance activities it performs, each time the RPV closure head is removed to detect and mitigate general degradation prior to returning the reactor to service. Additionally, the licensee stated that the threads in the RPV flange are inspected for damage, cleaned, and lubricated prior to reinstallation of the RPV studs.

3.4 Licensee's Proposed Alternative and Basis for Use

The licensee is proposing to eliminate the examination of threads in the RPV flanges required by Examination Category B-G-1, Item No. B6.40 of the ASME Code, Section XI, for the duration of the fifth 10-year ISI interval, or until the NRC approves an applicable alternative in NRC Regulatory Guide 1.147 or other document. The licensee's request is based on an evaluation by the Electric Power Research Institute (EPRI) documented in EPRI Technical Report No. 3002007626 (EPRI report), "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements," March 2016 (ADAMS Accession No. ML 16221A068). The licensee's submittals included information from the EPRI report regarding the generic stress analysis and the flaw tolerance evaluation, with additional plant-specific information to demonstrate applicability of the EPRI results to FitzPatrick. The submittals also included information from the EPRI report regarding operating experience and potential degradation mechanisms for threads in the RPV flanges.

4.0 NRC STAFF EVALUATION

The NRC staff reviewed the information in Relief Request I5R-04, Revision 0. The NRC staff focused its evaluation on the plant-specific applicability of the generic analyses contained in Section 6, "Stress Analysis and Flaw Tolerance Evaluation," of the EPRI report to FitzPatrick. Section 4, "Operating Experience," and Section 5, "Evaluation of Potential Degradation Mechanisms," of the EPRI report regarding operating experience and potential degradation mechanisms have already been accepted by the NRC staff as indicated in the safety evaluation dated June 26, 2017, enclosed in correspondence to Exelon for a relief request for 19 units (ADAMS Accession No. ML 17170A013).

4.1 Stress Analysis

Stresses were determined from the finite element method analyses and used as input into the flaw tolerance analysis. The licensee described maintenance activities it performs each time the RPV closure head is removed to detect and mitigate general degradation prior to returning the reactor to service. The licensee stated that the threads in the RPV flange are inspected for damage, cleaned, and lubricated prior to reinstallation of the RPV studs. The NRC staff considers these activities beneficial to flaw detection and that they could potentially reduce flaw initiation. Therefore, the conservative nature of the stress and flaw tolerance analyses is
verified periodically and maintained.

The NRC staff issued a safety evaluation dated January 26, 2017 (ADAMS Accession No. ML 17006A109), on similar alternative requests that used the generic stress analysis for Vogtle Electric Generating Plant, Units 1 and 2, and Joseph M. Farley Nuclear Plant, Unit 1. Therefore, the current evaluation focuses on the licensee's demonstration of plant-specific applicability of this generic stress analysis to the RPV flange threads for FitzPatrick. In Relief Request I5R-04, the licensee summarized its plant-specific information in Table 1, "Comparison of JAFNPP Plant Parameters to Bounding Values Used in Analysis," and Table 2, "RPV Flange Thread Geometry."

10 CFR 50.55a RELIEF REQUESTS: I5R-02, I5R-03, I5R-04

4.1.1 Evaluation of the Plant Parameters in Table 1 of Relief Request I5R-04

In Table 1 of its submittal for Relief Request I5R-04, the licensee provided information on six key FitzPatrick plant parameters: number of studs, stud nominal diameter, RPV inside diameter at stud hole, flange thickness at stud hole, design pressure, and preload stress. This table shows that the stud nominal diameter for FitzPatrick is the same as that in the generic stress analysis, and the preload stress for FitzPatrick is less than the corresponding generic value, indicating that these two parameters are bounded by the generic analysis. As a result, only the other four parameters need to be evaluated. Three of them are used to calculate the operating pressure load per stud through the following equation:

Load per stud= π(design pressure)(RPV inside diameter at stud hole)^2/(4xNo. of studs)

The NRC staff verified the licensee's calculation and confirmed that the load per stud for FitzPatrick is less than the corresponding generic value. Therefore, these three additional parameters are also bounded by the generic analysis. The last parameter (flange thickness at stud hole) leaves less RPV flange material in front of the critical crack front for FitzPatrick. Unfortunately, this is not evaluated by the licensee in the submittal.

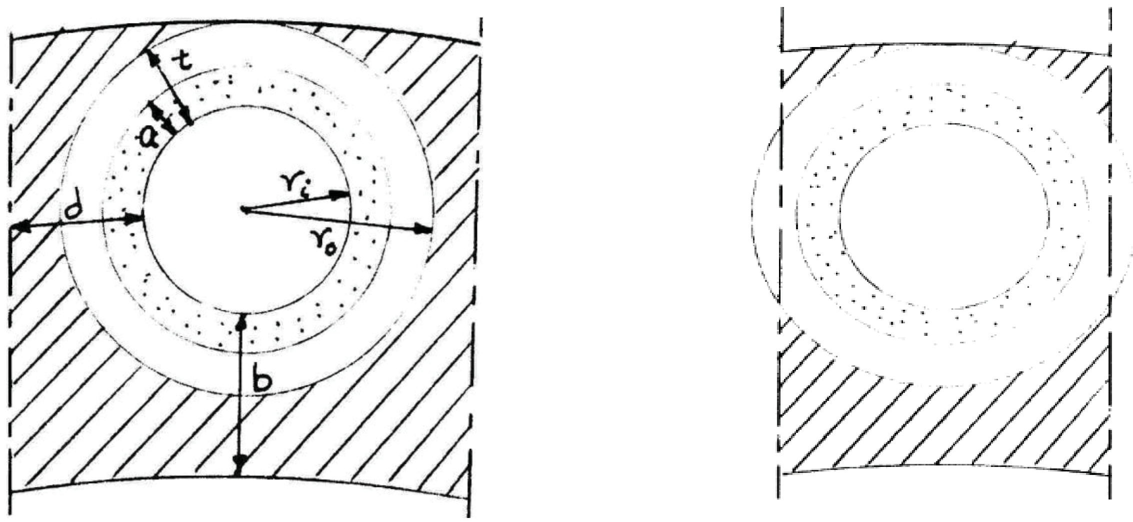
4.1.2 Evaluation of the Effect of a Seemingly Unbounded Parameter on Stresses

In Table 1 of its submittal for Relief Request 15R-04, the RPV flange thickness at the stud hole for FitzPatrick is 13.5 inches versus 16 inches for the generic stress analysis and is not bounded by the generic analysis. This feature is common to all boiling-water reactors (BWRs). However, due to an oversight, the unboundedness was not evaluated in the June 26, 2017, safety evaluation for the relief request for the 19 units, which included many BWRs. The NRC staff has evaluated the significance of this seemingly unbounded parameter in this relief request. This safety evaluation should be referenced in future applications for BWRs. The FitzPatrick RPV has 52 studs around the circumference with an inner RPV radius of 109.4 inches versus 54 studs and 86.5 inches for the generic analysis. This makes FitzPatrick's circumferential thickness between stud holes approximately twice that of the generic analysis. Therefore, for the same preload, FitzPatrick's axial stresses in the material between the stud holes will be much less than the generic analysis and are, therefore, bounded by it. For comparison, the NRC staff provides schematics of the generic stress model and the FitzPatrick model in Figure 1 below with key dimensions (approximate values) listed in Table 1. The above observation is different in the RPV radial direction where the total flange radial thickness (considering both sides of the stud hole) is 7.5 inches for FitzPatrick versus 9 inches for the generic analysis. Since the preload contributes the most to the maximum K (85 percent to 96 percent according to Figure 6-1 of the EPRI report), examining the stress plot for the preload case in Figure 6-5 of the EPRI report is sufficient. Figure 6-5 shows that at the assumed crack location (i.e., - 15 percent down from the top edge of the figure), the axial stress is high in areas adjacent to the stud hole, but decreases rapidly to low and then negative values toward the RPV flange outer edge. This means that the stress pattern of the generic analysis is not sensitive to the reduced thickness between the stud hole and the outer edge of the FitzPatrick RPV flange. This observation applies to the reduced thickness between the

10 CFR 50.55a RELIEF REQUESTS: I5R-02, I5R-03, I5R-04

stud hole and the inner edge of the FitzPatrick RPV flange, but of a much less concern because a significant portion of the thickness is under compressive stresses.

Figure 1. Schematics of the RPV Flange Stud Hole



FitzPatrick Generic

Table 1. Approximate Dimensions for Figure 1 for the FitzPatrick and Generic Models

	Relevant to Stress Model				Relevant to Fracture Model	
Key Dimensions	r_i (inches)	t (inches)	B (inches)	d (inches)	r_o (inches)	a (inches)
Generic	3.5	3.06	5.94	2.08	6.56	1.53
FitzPatrick	3	2.55	4.95	4.09	5.55	1.275

4.1.3 *Evaluation of the RPV Flange Thread Geometry in Table 2 of Relief Request I5R-04*

In Table 2 of its submittal for Relief Request I5R-04, the RPV flange thread geometry, which shows that for FitzPatrick, the pitch is eight threads per inch and the depth of threads is 0.06765 inch. In the generic stress analysis, the corresponding values are eight threads per inch and 0.06500 inch. The NRC staff evaluated differences of this magnitude in thread geometry on the final K results in the June 26, 2017, safety evaluation for the relief request for the 19 units and concluded that the impact is negligible. The same conclusion applies to FitzPatrick.

4.1.4 Loads and Resulting Stresses

The preload stress and the load per stud due to pressure for FitzPatrick are bounded by the generic analysis. The NRC staff found that the maximum heatup rate for FitzPatrick that is specified in Technical Specification 3.4.9, "RCS Pressure and Temperature (PIT) Limits," is also bounded by the generic heatup rate of 100 degrees Fahrenheit (°F)/hour. Please note that RCS PIT limits stands for reactor coolant system pressure-temperature limits. Therefore, all applied loads for FitzPatrick are bounded by the generic loads. Regarding the RPV flange stresses due to preload, the NRC staff's qualitative analysis indicated that flange axial stresses are not sensitive to the flange thickness at the stud hole. Based on its review, the NRC staff determined that the generic stress analysis results apply to FitzPatrick. However, since the driving force (i.e., the applied stress intensity factor, or the applied K) of the flaw tolerance analysis depends on the component geometry and the postulated flaw shape, the effect of the reduced RPV flange thickness on the flaw tolerance evaluation needs to be addressed. This is evaluated below in Section 4.2.

4.2. Flaw Tolerance Analysis

The licensee referenced the flaw tolerance analysis in the EPRI technical report as part of its basis to support the proposed alternative. The flaw tolerance analysis in the EPRI report, including the crack growth analysis, is based on the principles of linear elastic fracture mechanics. Similar to evaluation of the stress analysis, the NRC staff's current evaluation of the flaw tolerance analysis focuses on the effect to the generic analysis results due to the following FitzPatrick RPV flange information: (1) the flange material property and the bolt-up temperature and (2) the reduced flange thickness.

4.2.1 Evaluation of the Effect of the RPV Flange Material Property and the Bolt-Up Temperature on Applied K

In Table 4 of its submittal for Relief Request I5R-04, the licensee provided its RPV flange RT_{NDT} of 30 degrees Fahrenheit (°F) and bolt-up temperature of ≥ 60 °F for FitzPatrick. The NRC staff confirmed that "the information was obtained from plants as well as the NRC RVID2 database."

This information is consistent with that in the safety evaluation approving the relocation of FitzPatrick P/T limits from technical specifications to the licensee-controlled pressure temperature limits report in Amendment No. 292 for FitzPatrick, dated October 3, 2008 (ADAMS Accession No. ML082630365). Since preload is the dominant contributor to applied K, evaluation of the allowable Kat the lowest P/T limits temperature is appropriate. Applying the ($T-RT_{NDT}$) of 30 °F to the fracture toughness (K_{IC}) equation in ASME Code, Section XI, Appendix A, the NRC staff verified the licensee's calculated K_{IC} value of 71 ksi√ in. Applying the acceptance criteria of ASME Code, Section XI, IWB-3600 (with safety margin of √10), the NRC staff verified that the allowed applied K would be 22.45 ksi√in, which is greater than all maximum K values in Table 3 for the preload case of the generic analysis. Based on its review, the NRC staff determined that FitzPatrick is bounded by the generic flaw tolerance analysis.

4.2.2 Evaluation of the Effect of the Unbounded Parameter on Applied K

The NRC staff examined the generic RPV flange model schematics (Figures 6-2 and 6-8 of the EPRI report) and the FitzPatrick geometry features discussed in Section 4.2 above and determined that a thick-cylinder model (with r_i and r_o shown in Figure 1) with an inner circumferential crack of a uniform depth ("a" shown in Figure 1) under uniform axial stresses can be used to estimate the adjustment factor for the maximum applied K at the crack tip close to the RPV flange outer edge for FitzPatrick. This adjustment factor is needed to account for the geometric differences between the FitzPatrick model and the generic model. This estimation is conservative because, as indicated in Figure 1, much more flange material close to the RPV flange outer edge outside the imaginary thick-cylinder is not considered in the FitzPatrick thick-cylinder model than in the generic thick-cylinder model. The applied K for the simplified thick-cylinder model and the key geometric parameters are summarized in the following table for both the generic and FitzPatrick cases.

10 CFR 50.55a RELIEF REQUESTS: I5R-02, I5R-03, I5R-04

Table 2. Key Parameters for the Applied K Calculation

Case	r_i/r_o	a/t	Applied K [1]	Geometry Adjustment Factor
Generic	0.534	0.5	2.605xstress	1.0
FitzPatrick	0.54	0.5	2.378xstress	0.91

A lower preload stress and a geometry adjustment factor of 0.91 for the applied K for FitzPatrick means that the applied K for FitzPatrick is bounded by the generic flaw evaluation, even though the thickness between the stud hole and the RPV flange outer edge for FitzPatrick is smaller than the generic model.

Regarding use of the crack growth analysis in the EPRI report to support the proposed alternative, the NRC staff considers it acceptable because the assumption of 400 occurrences of preload and 4,000 occurrences for heatup/cooldown for an 80-year period in the generic analysis are conservative for FitzPatrick.

4.0 CONCLUSION

As set forth above, the NRC staff determines that for Relief Request I5R-04, Revision 0, the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for Relief Request I5R-04, Revision 0. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year ISI interval, which began on August 1, 2017, and is currently scheduled to end on June 15, 2027.

All other requirements of the ASME Code, Section XI, for which an alternative has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear lnservice Inspector. Any ASME Code, Section XI, RVI components that are not included in this relief request will continue to be inspected in accordance with the ASME Code, Section XI requirements.

Principal Contributor: Simon Sheng

Date: May 30, 2018

(1) Page 27.7, "The Stress Analysis of Cracks Handbook," Second Edition, Hiroshi Tada, Paul Paris, and George Irwin.

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF RELIEF REQUESTS FOR ALTERNATIVES TO CERTAIN ASME CODE REQUIREMENTS (CAC NOS. MG0116, MG0117, AND MG0118; EPID L-2017-LLR-0083, EPID L-2017-LLR-0084, AND EPID L-2017-LLR-0085) DATED MAY 30, 2018

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ADAMS Accession No.: ML18039A854 *by safety evaluation dated

OFFICE	NRR/DORL/LPL1/PM	NRR/DORL/LPL1/LA	NRR/DMLR/MVIB/BC(A)*
NAME	BVenkataraman	LRonewicz (JBurkhardt for)	SRuffin
DATE	02/08/2018	05/21/2018	01/29/2018
OFFICE	NRR/DMLR/MPHB*	NRR/DORL/LPL1/PM	NRR/DORL/LPL1/BC
NAME	DAlley	THood	JDanna
DATE	12/03/2017	05/16/2018	5/30/2018

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, 0.C. 20555-0001

September 10, 2018

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: JAMES A FITZPATRICK NUCLEAR POWER PLANT-ISSUANCE OF RELIEF FROM THE REQUIREMENTS OF THE ASME CODE N-702 FOR PLANT NOZZLE-TO-VESSEL WELDS AND INNER RADII EXAMINATIONS (EPID L-2017-LLR-0093)

Dear Mr. Hanson:

By letter dated September 29, 2017, as supplemented by letter dated March 2, 2018, Exelon Generation Company, LLC (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for relief from certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI requirements regarding the fifth 10-year inservice inspection program at the James A FitzPatrick Nuclear Power Plant (FitzPatrick).

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternatives on the basis that the alternatives provide an acceptable level of quality and safety.

This alternative is requested for the duration of the FitzPatrick fifth 10-year in service inspection interval beginning on August 1, 2017, and scheduled to end on June 15, 2027, and also for the remaining term of the FitzPatrick Renewed Operating License, which expires on October 17, 2034. Conditions are defined in NRC Regulatory Guide 1.147, Revision 17.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1).

If you have any questions, please contact the Project Manager, Tanya Hood, at 301-415-1387 or Tanya.Hood@nrc.gov.

Sincerely,

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosure:
Safety Evaluation

cc: Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELIEF REQUEST NO. I5R-05, TO UTILIZE ASME CODE CASE N-702

EXELON FITZPATRICK, LLC AND EXELON GENERATION COMPANY, LLC JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated September 29, 2017[1], as supplemented by letter dated March 2, 2018[2], Exelon Generation Company, LLC (Exelon, the licensee) submitted Relief Request I5R-05 to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), Section XI, Table IWB-2500-1, regarding the fifth 10-year inservice inspection (ISI) program at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick). This alternative is requested for the duration of the FitzPatrick fifth 10-year inservice inspection interval beginning on August 1, 2017, and scheduled to end on June 15, 2027, and also for the remaining term of the FitzPatrick Renewed Operating License, which expires on October 17, 2034.

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee proposes to use the inspection requirements documented in ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1." For the VT-1 visual examinations allowed by ASME Code Case N-702, the licensee proposes to use ASME Code Case N-648-1, "Alternative Requirements for Inner Radius Examination of Class 1 Reactor Vessel Nozzles, Section XI, Division 1," with associated required conditions specified in Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," dated August 2014[3]. The regulation at 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety.

2.0 REGULATORY REQUIREMENTS

Section 50.55a(g) of 10 CFR states, in part, that ISI of certain ASME Code Class 1, 2, and 3 systems and components be performed in accordance with Section XI of the ASME Code, except the design and access provisions and the preservice examination requirements, and

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1 Agencywide Documents Access and Management System (ADAMS) Accession No. ML17275A208
2 ADAMS Accession No. ML18064A278
3 ADAMS Accession No. ML13339A689

applicable addenda incorporated by reference in the regulations, as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained.

Section 50.55a(z) of 10 CFR states, in part, that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC prior to implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative 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.

The regulations require that inservice examination of components and system pressure tests conducted during the successive 120-month inspection intervals (following the initial 120-month inspection interval) must comply with the requirements in the latest edition and addenda of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(a) 12 months before the start of the 120-month interval (or the optional ASME Code Cases listed in RG 1.147, Revision 17, subject to the conditions listed in 50.55a(b)). The applicable ASME Code of record for the fifth 10-year ISI interval for FitzPatrick is ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

3.0 LICENSEE'S EVALUATION

3.1 Background

For all reactor pressure vessel (RPV) nozzle-to-vessel shell welds and nozzle inner radii, ASME Code, Section XI requires 100 percent inspection during each 10-year ISI interval. However, ASME Code Case N-702 provides an alternative, which reduces the inspection of RPV nozzle-to-vessel shell welds and nozzle inner radii areas from 100 percent to 25 percent of the nozzles for each nozzle type during each 10-year interval. This ASME Code Case was conditionally approved in RG 1.147, Revision 17. For application of ASME Code Case N-702, the licensee is required to address the conditions specified in RG 1.147, Revision 17, for ASME Code Case N-702:

> The applicability of Code Case N-702 must be shown by demonstrating that the criteria in Section 5.0 of NRC Safety Evaluation regarding BWRVIP-108 dated December 18, 2007[4] or Section 5.0 of NRC Safety Evaluation regarding BWRVIP-241 dated April 19, 2013[5] are met. The evaluation demonstrating the applicability of the Code Case shall be reviewed and approved by the NRC prior to the application of the Code Case.

BWRVIP-108, "BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," dated November 25, 2002,[6] and BWRVIP-241, "BWR Vessel and Internals Project, Probabilistic Fracture Mechanics [PFM] Evaluation for the Boiling Water Reactor

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4 ADAMS Accession No. ML073600374
5 ADAMS Accession No. ML13071A240
6 ADAMS Accession No. ML023330203

Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," dated April 26, 2011[7], contain PFM analysis results supporting ASME Code Case N-702. Both reports are for 40 years of operation. BWRVIP-241 contains additional PFM results supporting revision of the evaluation criteria under "Conditions and Limitations" in the safety evaluation (SE) for BWRVIP-108. The SE for BWRVIP-241 accepted the revised criteria.

The NRC staff issued a revised SE dated April 26, 2017[8], for license renewal for BWRVIP-241, Appendix A, "BWR Nozzle Radii and Nozzle-to-Vessel Welds Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)." This license renewal Appendix A extends the applicability of the BWRVP-108 and BWRVIP-241 methodologies, and, therefore, ASME Code Case N-702, from 40 years to the period of extended operation.

ASME Code Case N-702 allows that VT-1 visual examination may be performed in lieu of volumetric examination for Examination Item Number B3. 100 nozzle inner radius sections. ASME Code Case N-648-1, as conditionally accepted by RG 1.147, Revision 17, requires that nozzle inner radius examinations must use the allowable flaw length criteria of ASME B&PV Code Table IWB-3512-1 with limiting assumptions on the flaw aspect ratio.

3.2 ASME Code Component Affected

The affected components belong to Examination Category B-D, "Full Penetration Welded Nozzles in Vessels," under Examination Item Number B3.90, "Nozzle-to-Vessel Welds," and B3.100, "Nozzle Inside Radius Section."

Table 1

RPV Nozzle-to-Vessel Welds and Inner Radii Subject to this Request

Identification Number	Description	Total Number	Minimum Number to be Examined
N1	Recirculation Outlet	2	1
N2	Recirculation Inlet	10	3
N3	Main Steam Outlet	4	1
N5	Core Spray	2	1
N-TH-A/C	Closure Head Instrumentation	2	1
N-TH-B	Closure Head Vent	1	1
N8	Jet Pump Instrumentation	2	1

3.3 Applicable Code Edition and Addenda

This request applies to the fifth 10-year ISI interval and the remaining term of the FitzPatrick renewed operating license in which FitzPatrick adopted the 2007 Edition through the 2008 Addenda of ASME Code Section XI as the Code of Record.

[7] ADAMS Accession No. ML111190077
[8] ADAMS Accession No. ML17114A096

3.4 Applicable Code Requirements

ASME Section XI, Table IWB-2500-1, Examination Category B-D, requires a volumetric examination of all nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles each 10-year interval.

3.5 Licensee's Proposed Alternative

The licensee proposed to implement ASME Code Case N-702 and reduce the ASME Code required volumetric examinations for all RPV nozzle-to-shell welds and inner radii to a minimum of 25 percent of the nozzle inner radii and nozzle-to-shell welds, including at least one nozzle from each system and nominal pipe size during each inspection interval. The required examination volume for the reduced set of nozzles remains at 100 percent of that depicted in Figures IWB-2500-7 (a) through (d), as applicable in the ASME Code.

ASME Code Case N-702 stipulates that a VT-1 examination may be used in lieu of the volumetric examination for the inner radii. The licensee stated that if a VT-1 visual examination is performed, it will be consistent with ASME Code Case N-648-1 and the associated required conditions specified in RG 1.14 7, Revision 17.

3.6 Licensee's Basis for Alternative

The alternative is based on the PFM results documented in the BWRVIP-241 report. The licensee proposed that it met the evaluation criteria in the SE for BWRVIP-241 as follows:

(1) Max RPV Heatup/Cooldown Rate

The maximum RPV heatup/cooldown rate is limited to < 115°F (degrees Fahrenheit)/hour.

FitzPatrick Technical Specification Surveillance Requirement 3.4.9.1 reactor coolant system heatup and cooldown rates are s 100°F in any one hour period and thus meet the requirement of Condition 1.

(2) Recirculation Inlet (N2) Nozzles

$(pr/t)/C_{RPV} < 1.15$, where

p = RPV normal operating pressure (psi), r = RPV inner radius (inch),
t = RPV wall thickness (inch), and
$C_{i\text{-}RPV}$ = Inlet RPV constant
$C_{i\text{-}RPV}$ = 19332.

The FitzPatrick result based on the input parameters for this nozzle per the licensee submittal is (pr/t) /CRPv = 0.86 ([(1040)(110.375)/6.875]/19332), thus meeting the requirements of Condition 2.

(3) Recirculation Inlet (N2) Nozzles

$[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]/C_{NozzLE} \leq 1.47$, where

p = RPV normal operating pressure
r_o = nozzle outer radius (inch),
r_i = nozzle inner radius (inch), and
$C_{i\text{-}NOZZLE}$ = Inlet nozzle constant
$C_{i\text{-}NOZZLE}$ = 1637.

The FitzPatrick result based on the input parameters for this nozzle per the licensee submittal is $[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]/C_{i\text{-}NOZZLE} = 1.37$ ($[1040(10.22^2 + 6.19^2)/(10.22^2 - 6.188^2)]/1637$), thus meeting the requirements of Condition 3.

(4) Recirculation Outlet (N1) Nozzles

$(pr/t)/C_{RPV} \leq 1.15$, where

r = RPV inner radius (inch),
t = RPV wall thickness (inch), and
$C_{o\text{-}RPV}$ = Outlet RPV constant
$C_{o\text{-}RPV}$ = 16171.

The FitzPatrick result based on the input parameters for this nozzle per the licensee submittal is $(pr/t)/C_{o\text{-}RPV} = 1.03$ ($[(1040)(110.375)/6.875]/16171$), thus meeting the requirements of Condition 4.

(5) Recirculation Outlet (N1) Nozzles

$[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]/C_{o\text{-}NOZZLE} \leq 1.59$, where

r_o = nozzle outer radius (inch),
r_i = nozzle inner radius (inch), and
$C_{o\text{-}NOZZLE}$ = Outlet nozzle constant
$C_{o\text{-}NOZZLE}$ = 1977.

The FitzPatrick result based on the input parameters for this nozzle per the licensee submittal is $[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]/C_{o\text{-}NOZZLE} = 1.08$ ($[1040(21.66^2 + 12.69^2)/(21.66^2 - 12.69^2)]/1977$), thus meeting the requirements of Condition 5.

3.7 Duration of the Proposed Alternative

This alternative is requested for the duration of the FitzPatrick fifth 10-year ISI interval beginning on August 1, 2017, and scheduled to end on June 15, 2027, and for the remaining term of the FitzPatrick Renewed Operating License, which expires on October 17, 2034.

4.0 NRC STAFF EVALUATION

The NRC staff reviewed the information in Relief Request I5R-05, including the supplemental information in the letter dated March 2, 2018. The NRC staff reviewed the status of the referenced BWRVIP reports and found application of the referenced BWRVIP reports to be acceptable, provided that the NRC conditions associated with the latest safety evaluation for each BWRVIP report are implemented.

The licensee proposed an alternative to implement ASME Code Case N-702 for all FitzPatrick RPV nozzle-to-vessel shell penetration welds and nozzle inner radii using the criteria in BWRVIP-241. The applicability of the BWRVIP-241 report to an ASME Code Case N-702 alternative is demonstrated by showing that Criteria 2 through 5 within Section 5.0 of the NRC SE for BWRVIP-241 are met for the bounding nozzles, and that Criterion 1 is met for all components included in the proposed alternative.

The NRC staff confirms that Criterion 1 is satisfied because FitzPatrick Technical Specification Surveillance Requirement 3.4.9.1 limits the maximum heatup/cooldown rate to less than or equal to 100°F/hour, well below the 115 °F/hour criterion limit.

For Criteria 2 to 5, the licensee provided plant-specific data and its evaluation of the driving force factors, or ratios, using the criteria established in Section 5.0 of the BWRVIP-241 SE. The NRC staff reviewed the licensee's calculations and confirms that they show that Criteria 2 to 5 are satisfied. Therefore, the BWRVIP-241 report is applicable, and the basis for using ASME Code Case N-702 is demonstrated for the FitzPatrick RPV nozzle-to-vessel welds and inner radii listed in Table 1 above.

By e-mail dated February 2, 2018[9], the NRC staff issued a request for additional information (RAI) requesting the licensee to report the probability of failure (PoF) values for low temperature overpressure (LTOP) and normal operation or discuss how the PoF values for LTOP are more limiting than those for normal operation. In its RAI response dated March 2, 2018, the licensee provided a plant-specific PFM analysis to supplement the criteria of ASME Code Case N-702 in order to demonstrate that the PoF remains acceptable over the period of extended operation.

The evaluation concluded the maximum PoF for an LTOP event is 3.0×10^{-9} per year, and the maximum PoF for normal operation is less than 8.0×10^{-9} per year. These PoFs are approximately 3 orders of magnitude lower than the acceptance criterion of 5×10^{-6} per year. Based on its review, the NRC staff finds the licensee's evaluation acceptable because the PoF due to either LTOP or normal operation is less than the NRC safety goal of 5×10^{-5} per year.

For the Examination Item Number B3.100 nozzle inner radius sections, the NRC staff finds the licensee's proposal to perform VT-1 visual examination in lieu of ultrasonic examination to be acceptable since the licensee will comply with ASME Code Case N-648-1, with associated required conditions specified in RG 1.147, Revision 17.

This alternative is requested for the duration of the FitzPatrick fifth 10-year ISi interval beginning on August 1, 2017, and scheduled to end on June 15, 2027, and for the remaining term of the FitzPatrick renewed operating license, which expires on October 17, 2034. The ASME Code Case N-702 examination requirements shall be met during this entire timeframe. Specifically, a minimum of 25 percent of nozzle inner radii and nozzle-to-shell welds, including at least one nozzle from each system and nominal pipe size, as identified in Relief Request I5R-05, will be examined during each 120-month ISI inspection interval in accordance with the conditions for the implementation of ASME Code Case N-702. These conditions are defined in RG 1.147, Revision 17. The licensee shall adhere to these requirements during both of the remaining

9 ADAMS Accession No. ML18033A139

FitzPatrick 10-year ISI program intervals, which span the balance of the FitzPatrick period of extended operation.

As part of its review, the NRC staff considered the duration of the request (i.e., for the remainder of the licensee's current license). The NRC staff finds the duration of the request to be acceptable because the request lies within the licensee's current license; the request is limited to demonstrating that the conditions placed on NRC approved ASME Code Case N-702 are met, and none of the inputs or calculations required to meet the NRC conditions change with time.

5.0 CONCLUSION

As set forth above, the NRC staff determines that for Relief Request 1 SR-05, the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for Relief Request 1 SR-05. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year ISI interval and the remaining term of the FitzPatrick Renewed Operating License for ASME Category B-D, Item Numbers B3.90 and B3.100, until October 17, 2034.

The NRC staff notes that if the licensee intends to take exceptions to, or deviations from, the NRC staff-approved 8WRVIP inspection guidelines, this will require the licensee to revise and resubmit this relief request. The licensee shall obtain NRC staff approval for such exceptions prior to implementing the revised inspection guidelines for FitzPatrick pursuant to 10 CFR 50.55a(z).

All other requirements of the ASME Code, Section XI, for which an alternative has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector. Any ASME Code, Section XI RVI components that are not included in this relief request will continue to be inspected in accordance with the ASME Code, Section XI requirements.

Principal Contributor: C. Fairbanks

Date: September 10, 2018

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT—ISSUANCE OF RELIEF FROM THE REQUIREMENTS OF THE ASME CODE N-702 FOR PLANT NOZZLE-TO-VESSEL WELDS AND INNER RADII EXAMINATIONS (EPID L-2017-LLR-0093) DATED SEPTEMBER 10, 2018

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CORRECTION TO SAFETY EVALUATION
BY

THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED ALTERNATIVE TO UTILIZE ASME CODE CASE N-789-1

EXELON FITZPATRICK,
LLC

EXELON GENERATION COMPANY,
LLC

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-
333

1.0 INTRODUCTION

By letter dated May 4, 2017 (Agencywide Documents Access and Management System Accession No. ML17124A303), Exelon Generation Company, LLC (Exelon, the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, IWA-4000, for the repair of Class 2 and 3 moderate energy carbon steel raw water service system piping at the James A. Fitzpatrick Nuclear Power Plant (FitzPatrick).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee proposed an alternative to use ASME Code Case N-789-1, "Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping for Raw Water Service, Section XI, Division 1," for the repair of the cooling water system piping on the basis that complying with the specified ASME Code requirement to repair the subject piping would result in hardship and/or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

Article IWA-4400 of the ASME Code, Section Xl, requires that unacceptable flaws in ASME Code Class 2 and 3 components be corrected by repair or replacement activity or be accepted by supplemental examination and flaw evaluation.

Adherence to Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4), which states, in part, that ASME Code Class 1, 2, and 3 components (including supports) will meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI.

The regulation in 10 CFR 50.55a(z) states, in part, that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the NRC, if the licensee demonstrates that (1) the proposed alternative provides 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.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components Affected

The affected components are ASME Code Class 2 and 3 moderate energy, carbon steel, raw water piping systems. Raw water is defined as water such as river, lake, well, or brackish/salt water used in plant equipment, area coolers, and heat exchangers. Moderate energy is defined as less than or equal to 200 degrees Fahrenheit (°F) (93 degrees Celsius (°C)) and less than or equal to 275 pounds per square inch gauge (psig) (1.9 MPa) maximum operating conditions.

3.2 ASME Code Edition and Addenda

The applicable Code of record for the fifth 10-year inservice inspection interval (ISI) at FitzPatrick is ASME Code Section XI, 2007 Edition through the 2018 Addenda.

3.3 ASME Code Requirements

ASME Code, Section XI, IWA-4000 provides requirements for welding, brazing, metal removal, and installation of repair/replacement activities.

3.4 Licensee's Reason for Request

In accordance with 10 CFR 50.55a(z)(2), Exelon requested a proposed alternative from the requirement for replacement or internal weld repair of wall thinning conditions resulting from degradation in Class 2 and 3 moderate energy carbon steel raw water piping systems in accordance with IWA-4000. Such degradation may be the result of mechanisms such as erosion, corrosion, cavitation, and pitting, but excluded are conditions involving flow-accelerated corrosion, corrosion-assisted cracking, or any other form of cracking. IWA-4000 requires repair or replacement in accordance with the owner's requirements and the original or later Construction Code. Other alternative repair or evaluation methods are not always practicable because of wall thinness and/or moisture issues. The primary reason for this request is to permit installation of a technically sound temporary repair to provide adequate time for evaluation, design, material procurement, planning, and scheduling of appropriate, permanent repair or replacement of the defective piping, considering the impact on system availability, maintenance rule applicability, and availability of replacement materials.

Performing Code repair/replacement in lieu of implementing this relief request would, in some cases, necessitate extending technical specification actions to install a permanent repair/replacement, putting the plant at higher safety risks compared with the short time necessary to install a technically sound pad repair. Use of N-789-1 may avoid a plant shutdown in situations where it may be necessary to shut the plant down for a Code repair/replacement activity. This could result in an unnecessary plant transient and the loss of safety system availability, as compared to maintaining the plant online.

Use of Code Case N-789-1 during refueling outages will enable a greater number of scheduled corrosion inspections during the outages. The ability to install non-intrusive repair pads rather than scheduling contingency plans for piping replacement will enable longer corrosion inspection windows, increased scope of inspection, and improved overall plant safety.

3.5 Licensee's Proposed Alternative and Basis for Use

In accordance with 10 CFR 50.55a(z)(2), Exelon proposed to implement the requirements of ASME Code Case N-789-1 ("Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping for Raw Water Service, Section XI, Division 1") as a temporary method to repair degradation in Class 2 and 3 moderate energy raw water piping systems resulting from mechanisms such as erosion, corrosion, cavitation, or pitting, but excluding conditions involving flow-accelerated corrosion, corrosion-assisted cracking, or any other form of cracking. These types of defects are typically identified by small leaks in the piping system or by preemptive non-Code required examinations performed to monitor the degradation mechanisms.

The alternative repair technique described in Code Case N-789-1 involves the application of a metal reinforcing pad welded to the exterior of the piping system, which reinforces the weakened area and restores pressure integrity. This repair technique will be utilized when it is determined that this temporary repair method is suitable for the particular defect or degradation being resolved.

Code Case N-789-1 requires that the cause of the degradation be determined and that the extent and rate of degradation in the piping be evaluated to ensure that there are no other unacceptable locations within the surrounding area that could affect the integrity of the repaired piping. The area of evaluation will be dependent on the degradation mechanism present. A baseline thickness examination will be performed for a completed structural pad, attachment welds, and surrounding area, followed by monthly thickness monitoring for the first 3 months, with subsequent frequency based on the results of this monitoring, but at a minimum of quarterly. Areas containing pressure pads shall be visually observed at least once per month to monitor for evidence of leakage. If the areas containing pressure pads are not accessible for direct observation, then monitoring will be accomplished by visual assessment of surrounding areas or ground surface areas above pressure pads on buried piping or monitoring of leakage collection systems, if available.

For the pressure pad design, the higher of two times the actual measured corrosion rate, or four times the estimated maximum corrosion rate, for the system will be used. If the actual measured corrosion rate in the degraded location is unavailable, the estimated maximum corrosion rate for the system assumed in the design will be calculated based on the same degradation mechanism as the degraded location.

Paragraph 3.2(i) of the Code Case includes an incorrect reference to NC-2650 for the flexibility analysis associated with Class 2 designs. The correct reference should be NC-3650. Exelon will comply with NC-3650.

The repair will be considered to have a maximum service life of the time until the next refueling outage when a permanent repair or replacement must be performed. Additional requirements for design of reinforcement pads, installation, examination, pressure testing, and inservice monitoring are provided in Code Case N-789-1.

Based on the above justification, the use of Code Case N-789-1 as a proposed alternative to the requirements of ASME Section XI will provide an acceptable level of quality and safety that does not impose an undue hardship. All other ASME Section XI requirements for which relief was not specifically requested and authorized by the NRC staff will remain applicable, including third party review by the Authorized Nuclear lnservice Inspector.

Code Case N-789-1 has not been incorporated into NRC Regulatory Guide 1.147, "lnservice Inspection Code Case Acceptability, ASME Section XI, Division 1," and thus, is not available for application at nuclear power plants without specific NRC approval. Therefore, Exelon requests use of this alternative repair technique described in the Code Case by this relief request.

3.6 Duration of Proposed Alternative

The proposed alternative is for use of the Code Case for the remainder of the 10-year inspection interval as specified in Section 2.0 of the licensee's May 4, 2017, letter. When Code Case N-789-1 is approved for use by the NRC, this relief request will no longer be applied, and the Code Case, including Regulatory Guide 1.147 conditions, will be used in lieu of this relief request. Any reinforcing pads installed before the end of the 10-year ISI interval will be removed during the next refueling outage, even if that refueling outage occurs after the end of the 10-year interval.

3.7 NRC Staff Evaluation of the Alternative

The NRC staff evaluated the adequacy of the proposed alternative in maintaining the structural integrity of the repaired subject piping. The staff focused on the following key elements of the proposed alternative to use Code Case N-789-1: (1) general requirements, (2) initial evaluation, (3) design requirements, (4) water-back application, (5) installation, (6) examination, (7) pressure testing, (8) inservice monitoring, and (9) hardship justification.

The NRC staff notes that many requirements specified in Code Case N-789-1 are not discussed in this safety evaluation, but they should not be considered as less important. As part of the NRC-approved proposed alternative, all requirements in the Code Case must be followed. Any exceptions to the Code Case that are approved in this safety evaluation also need to be followed.

3.7.1 General Requirements

The NRC staff notes that the proposed alternative requires the reinforcing pad be applied in accordance with a repair/replacement plan satisfying the requirements of the ASME Code, IWA-

4150. The design, materials, and installation requirements of the Construction Code and IWA-4000, except as stated in the Code Case, must be satisfied.

Code Case N-789-1 includes the following limitations: (1) the repair cannot be applied if the minimum required thickness of reinforcing pad necessary to satisfy the requirements of Section 3 of the Code Case is greater than the nominal thickness for the size and schedule of the piping; (2) additional reinforcement or repair on top of an existing reinforcing pad is prohibited; (3) reinforcing pads, including those installed during a refueling outage, shall not remain in service beyond the end of the next refueling outage; and (4) the repair is only applicable to piping not required to be ultrasonically examined for ISI.

The NRC staff finds that the proposed general requirements, including limitations, are appropriate and, therefore, acceptable.

3.7.2 Initial Evaluation

The NRC staff finds that the proposed initial evaluation in Code Case N-789-1 is acceptable because: (1) prior to installing the reinforcing pad, the proposed alternative requires that the base metal be ultrasonically examined to determine the cause and rate of degradation; (2) if the cause of damage is determined to be flow-accelerated corrosion, corrosion-assisted cracking, or any other form of cracking, the licensee will not use this Code Case to repair the subject piping; and (3) the proposed alternative requires an inspection be performed to determine the condition of the subject piping.

3.7.3 Design Requirements

The licensee stated that paragraph 3.2(i) of Code Case N-789-1 includes an incorrect reference to NC-2650 for the flexibility analysis associated with Class 2 designs. The correct reference should be NC-3650, and the licensee stated that it will comply with NC-3650. The NRC staff finds that the reference to NC-2650 in paragraph 3.2(i) of Code Case N-789-1 is incorrect; the correct reference is NC-3650, as stated by the licensee. Therefore, the staff finds the licensee's use of NC-3650 in lieu of NC-2650 to be acceptable.

The NRC staff finds that the reinforcing pads will be designed in accordance with the applicable requirements of the Construction Code or the ASME Code, Section III (NC-3100; ND-3100; NC-3600; and ND-3600, including Appendix II). The NRC staff notes that the proposed alternative clearly defines the pressure pads and structural pads such that each type of pad will be applied for specific pipe degradation and purpose.

The NRC staff notes that Code Case N-789-1, paragraph 3.1(a)(1), specifies that a pressure pad is designed with a corrosion rate of either two times the actual measured corrosion rate in that location or four times the estimated maximum corrosion rate for the system. The licensee stated that if the actual measured corrosion rate in the degraded location is unavailable, the estimated maximum corrosion rate for the system assumed in the design will be calculated based on the same degradation mechanism as the degraded location.

For the structural pad, the corrosion rate will be based on paragraph 3.2(f) in the Code Case, which requires that the predicted maximum degradation of the reinforced piping until the next refueling outage be included in the design. The predicted degradation of the piping will be based on in-situ inspection of, and established data for, similar base metals in similar environments. The proposed alternative requires that if the reinforcing pad is predicted to become exposed to the raw water, the predicted degradation of the reinforcing pad shall be based upon established data for base metals or weld metals with similar chemical composition to that used for the reinforcing pad.

The NRC staff notes that the Code Case does not provide a specific corrosion rate determination for the structural pad. It is not clear to the staff that the corrosion rate used in the structural pad design would be bounding other than by the fact that the structural pad will be designed for the duration until the next refueling outage. As a compensatory measure, the proposed alternative does require inservice monitoring to ensure the structural integrity of the repaired pipe using a structural pad. In addition, the proposed repair is limited to a maximum duration of one operating cycle. This relatively short duration of application should limit the degradation. However, should the actual corrosion rate exceed the projected corrosion rate during the operating cycle, and a leak develop at or around the installed pad, the proposed inservice monitoring will be able to detect such leakage, and the operator will be able to take corrective action.

The NRC staff notes that by the next refueling outage, the structural pad will be designed with partial penetration attachment welds that extend for a distance in each direction beyond the area predicted to infringe upon the required thickness. Final configuration of the structural pad (including attachment welds) will permit the examinations and evaluations required herein, including any required preservice or inservice examinations of encompassed or adjacent welds. The proposed alternative requires that the thickness of the reinforcing pad be sufficient to maintain required thickness until the next refueling outage.

Despite some concern about the corrosion rate used in the structural pad design, the NRC staff finds that the proposed alternative will provide reasonable assurance of the structural integrity and leakage integrity of the repaired piping until the next refueling outage because: (1) the structural pad will be designed to maintain required thickness until the next refueling outage, and (2) the proposed alternative requires periodic inservice monitoring as discussed further in this safety evaluation. Therefore, the NRC staff finds the aforementioned design requirements to be acceptable.

3.7.4 Water-Backed Applications

The proposed alternative requires the use of the shielded metal arc welding process with low-hydrogen electrodes for the attachment welds on water-backed piping. The proposed alternative further requires precaution be taken when welding a reinforcing pad to a leaking area. For piping materials other than P-No. 1, Group 1, the proposed alternative requires a surface examination that is to be performed no sooner than 48 hours after completion of welding. The NRC staff notes that waiting 48 hours after welding ensures that if delayed hydrogen cracking were to occur, it would be detected during the surface examination. Therefore, the NRC staff finds the proposed requirements for water-backed application to be acceptable.

3.7.5 Installation

The NRC staff finds that the proposed alternative requires the use a qualified welding procedure in accordance with the ASME Code, Section IX, and the Construction Code, in addition to requirements specified in the Code Case. Therefore, the NRC staff finds the proposed installation requirements to be acceptable.

3.7.6 Examination

The proposed alternative requires a surface examination (liquid penetrant or magnetic particle) and volumetric examination be performed of the pad, weld, and base metal after the reinforcing pad is welded to the pipe in accordance with Section III of the ASME Code or the Construction Code. The NRC staff finds the proposed acceptance examination follows Section III of the ASME Code and the Construction Code. Therefore, the staff finds the proposed acceptance examinations to be acceptable.

3.7.7 Pressure Testing

The proposed alternative requires that a system leakage test will be performed in accordance with IWA-5000 prior to, or as part of, returning the system to service. In addition, reinforcing pads attached to piping that have not been breached shall be equipped with pressure taps for performance of pressure testing. The NRC staff finds that the proposed pressure testing is acceptable because it is consistent with IWA-5000 of the ASME Code, Section XI.

3.7.8 Inservice Monitoring

For the structural pad, the proposed alternative requires that the pad be examined using ultrasonic or direct thickness measurement to record the thickness of the plate; the thickness at the attachment welds, including the underlying base metal; and, to the extent examinable in a 3-inch wide band, the thickness surrounding the repair as a baseline for subsequent monitoring of the repair. The licensee will monitor the structural pad monthly for the first quarter. The subsequent frequency will be based on the results of the monitoring activities, but at least quarterly.

For the pressure pad, the proposed alternative requires that the areas containing the pad be visually examined monthly for evidence of leakage. If the areas containing the pressure pad are not accessible for direct observation, the licensee will observe surrounding areas or ground surface areas above pressure pads on buried piping or leakage collection systems, if available.

The licensee stated that if the results of the monitoring program identify leakage or indicate that the structural margins required by the Code Case will not be maintained until the next refueling outage, the pad will be removed, and repair/replacement activities shall be performed prior to encroaching on the design limits.

The NRC staff finds that the proposed inservice monitoring requirements are acceptable because: (1) the frequency and the examination method are adequate to monitor the structural integrity of the pressure pad and structural pad, and (2) the acceptance criteria for the pressure pad and structural pad are clearly defined and adequate.

3.7.9 Applicable Duration

The licensee requested to use the proposed alternative for the fifth 10-year inspection interval or until such time that Code Case N-789-1 is approved for use by the NRC. The fifth 10-year ISI interval at FitzPatrick began on June 16, 2017, and is scheduled to end on June 15, 2027. The licensee clarified that any reinforcing pads installed before the end of the 10-year ISI interval will be removed during the next refueling outage, even if that refueling outage occurs after the end of the 10-year interval. Installed reinforcing pads are designed to support a maximum of one cycle of operation from one refueling outage to the next refueling outage. The NRC staff finds that installed reinforcing pads are acceptable to remain in service beyond the end date of the 10-year lSI interval if that interval end date falls midcycle and if the pad is removed in the next scheduled refueling outage.

3.7.10 Hardship Justification

The NRC staff finds that performing a plant shutdown to repair the subject piping would cycle the unit and increase the potential of an unnecessary transient, resulting in undue hardship. Additionally, performing the ASME Code repair during normal operation could necessitate extending technical specification actions, thus placing the plant at higher safety risk than warranted. Therefore, the NRC staff determines that compliance with the specified ASME Code repair requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.8 Summary

The NRC staff finds that the proposed alternative will provide reasonable assurance of the structural integrity and leaktightness of the repaired cooling water system pipe because: (1) the scope of the application is clearly defined; (2) the pressure pad and structural pad will be designed in accordance with the Construction Code and ASME Code, Section III, and specific requirements as specified in Code Case N-789-1; (3) the degraded pipe will be examined and evaluated prior to the repair; (4) acceptance examinations will be performed to verify the condition of the repair; (5) the inservice monitoring will be performed to verify the pipe wall thickness and potential degradation; and (6) pressure testing will be performed in accordance with IWA-5000 of the ASME Code, Section XI.

4.0 CONCLUSION

As set forth above, the NRC staff finds that complying with IWA-4000 of the ASME Code, Section XI, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The staff finds that the licensee demonstrated its proposed alternative to use Code Case N-789-1 will provide reasonable assurance that the structural integrity and leakage integrity of the subject cooling water system piping will be maintained. 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). The NRC staff authorizes the proposed alternative as documented in the submittal dated May 4, 2017, for the temporary repair of Class 2 and 3 moderate energy carbon steel raw water service piping at FitzPatrick for the fifth 10-year inspection interval.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and authorized by NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Robert Davis

Date: December 12, 2017
Correction Date: January 3, 2018

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

December 11, 2017

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – ISSUANCE OF RELIEF REQUEST-ALTERNATIVE TO CERTAIN REQUIREMENTS OF THE ASME CODE REGARDING USE OF ASME CODE CASE N-513-4 (CAC NO. MF9641; EPIO L-2017-LLR-0023)

Dear Mr. Hanson:

By letter dated April 20, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 17110A274), Exelon Generation Company, LLC (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for the use of an alternative to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements at the James A. FitzPatrick Nuclear Power Plant.

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(2), the licensee requested to use the alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. The proposed alternative would allow the licensee to use ASME Code Case N-513-4, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping, Section XI, Division 1," for the evaluation and temporary acceptance of flaws in moderate energy Class 2 and 3 piping, in lieu of specified ASME Code requirements.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the proposed alternative provides reasonable assurance of structural integrity of the subject components. The staff concludes that the licensee has adequately addressed all of the regulatory requirements, set forth in 10 CFR 50.55a(z)(2). Accordingly, the NRC staff authorizes the use of the licensee's proposed alternative, as described in its April 20, 2017, letter, to use ASME Code Case N-513-4 at FtizPatrick for the fifth 10-year inservice inspection interval, which began June 16, 2017, and is scheduled to end on June 15, 2027, or until such time as the NRC approves Code Case N-513-4 for general use through revision of Regulatory Guide 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," or another document.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and approved by NRC staff in this proposed alternative remain in effect.

If you have any questions, please contact the Project Manager, Booma Venkataraman, at 301-415-2934 or Booma.Venkataraman@nrc.gov.

Sincerely,

James G. Danna, Chief

Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosure:
Safety Evaluation

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200 Exelon Way
Kennett Square, PA 19348

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May 30, 2018

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Braidwood Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-37 andNPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

James A. FitzPatrick Nuclear Power Plant
Renewed Facility Operating License No. DPR-59
NRC Docket No. 50-333

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Limerick Generating Station, Units 1 and 2

Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Nine Mile Point Nuclear Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-63 and NPF-69
NRC Docket Nos. 50-220 and 50-410

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

R.E. Ginna Nuclear Power Plant
Renewed Facility Operating License No. DPR-18
NRC Docket No. 50-244

Three Mile Island Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Proposed Alternative to Utilize Code Case N-879

In accordance with 10 CFR 50.55a(z)(2), Exelon Generation Company, LLC (Exelon) is requesting a proposed alternative to 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 compliance with the code results in hardship without a compensating increase in quality. Specifically, this proposed alternative concerns the use of Code Case N-879, "Use of Micro-Alloyed Carbon Steel Bar in Patented Mechanical Joints and Fittings, Classes 1, 2, and 3 Section III, Division 1." This Code Case allows the use of a material that does not comply with the limitations on material specifications and grades mandated by ASME Section III. A separate relief request being submitted under a separate cover, requests approval of Code Cases N-878 and N-880 to allow the procurement of material from a material supplier that does not possess ASME accreditation as a Quality System Certificate Holder or an NPT Certificate Holder.

There are no regulatory commitments contained in this letter. Exelon requests your review and approval of this fleet request by May 30, 2019.

If you have any questions, please contact Tom Loomis (610) 765-5510

Respectfully,

James Barstow
Director – Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachment: Proposed Alternative to Utilize Code Case N-879

cc: Regional Administrator - NRC Region I
Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Braidwood Station
NRC Senior Resident Inspector - Byron Station
NRC Senior Resident Inspector - Calvert Cliffs Nuclear Power Plant
NRC Senior Resident Inspector - Clinton Power Station
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - James A. FitzPatrick Nuclear Power Plant
NRC Senior Resident Inspector - LaSalle County Station
NRC Senior Resident Inspector - Limerick Generating Station
NRC Senior Resident Inspector - Nine Mile Point Nuclear Station
NRC Senior Resident Inspector - Peach Bottom Atomic Power Station
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
NRC Senior Resident Inspector - R. E. Ginna Nuclear Power Plant
NRC Senior Resident Inspector - Three Mile Island Nuclear Station, Unit 1
NRC Project Manager - Braidwood Station
NRC Project Manager - Byron Station
NRC Project Manager - Calvert Cliffs Nuclear Power Plant
NRC Project Manager - Clinton Power Station
NRC Project Manager - Dresden Nuclear Power Station
NRC Project Manager - James A. FitzPatrick Nuclear Power Plant
NRC Project Manager - LaSalle County Station
NRC Project Manager - Limerick Generating Station
NRC Project Manager - Nine Mile Point Nuclear Station
NRC Project Manager - Peach Bottom Atomic Power Station
NRC Project Manager - Quad Cities Nuclear Power Station
NRC Project Manager - R.E. Ginna Nuclear Power Plant
NRC Project Manager - Three Mile Island Nuclear Station, Unit 1

Attachment
Proposed Alternative to Utilize Code Case N-879

Proposed Alternative to Utilize Code Case N-879 in Accordance with 10 CFR 50.55a(z)(2)

1. ASME Code Component(s) Affected:

All ASME Class 1, 2, and 3 carbon steel piping systems Nominal Pipe Size (NPS) 2 and smaller.

2. Applicable Code Edition and Addenda:

PLANT	INTERVAL	EDITION	START	END
Braidwood Station, Units 1 and 2	Fourth	2013 Edition	August 29, 2018 October 17, 2018	July 28, 2028 October 16, 2028
Byron Station, Units 1 and 2	Fourth	2007 Edition, through 2008 Addenda	July 16, 2016	January 15, 2026
Calvert Cliffs Nuclear Power Plant, Units 1 and 2	Fourth	2004 Edition	October 10, 2009	June 30, 2019
Clinton Power Station, Unit 1	Third	2004 Edition	July 1, 2010	June 30, 2020
Dresden Nuclear Power Station, Units 2 and 3	Fifth	2007 Edition, through 2008 Addenda	January 20, 2013	January 19, 2023
James A. FitzPatrick Nuclear Power Plant	Fifth	2007 Edition, through 2008 Addenda	August 1, 2017	June 15, 2027
LaSalle County Stations, Units 1 and 2	Fourth	2007 Edition, through 2008 Addenda	October 1, 2017	September 30, 2027
Limerick Generating Station, Units 1 and 2	Fourth	2007 Edition, through 2008 Addenda	February 1, 2017	January 31, 2027
Nine Mile Point Nuclear Station, Unit 1	Fourth	2004 Edition	August 23, 2009	August 22, 2019

PLANT	INTERVAL	EDITION	START	END
Nine Mile Point Nuclear Station, Unit 2	Third	2004 Edition	April 5, 2008	August 22, 2018
Nine Mile Point Nuclear Station, Unit 2	Fourth	2013 Edition	August 23, 2018	August 22, 2028
Peach Bottom Atomic Power Station, Units 2 and 3	Fourth	2001 Edition, through 2003 Addenda	November 5, 2008	December 31, 2018
Peach Bottom Atomic Power Station, Units 2 and 3	Fifth	2013 Edition	January 1, 2019	December 31, 2028
Quad Cities Nuclear Power Station, Units 1 and 2	Fifth	2007 Edition, through 2008 Addenda	April 2, 2013	April 1, 2023
R.E. Ginna Nuclear Power Plant	Fifth	2004 Edition	January 1, 2010	December 31, 2019
Three Mile Island Nuclear Station, Unit 1	Fourth	2004 Edition	April 20, 2011	April 19, 2022

3. Applicable Code Requirement:

ASME Code, Section III, NB/NC/ND-2121 (a), of the 1971 Edition through the 2017 Edition, provides requirements for materials to be used in Class 1, 2, and 3 piping systems.

4. Reason for Request:

In accordance with 1 O CFR 50.55a(z}(2), Exelon Generation Company, LLC (Exelon) is requesting a proposed alternative from the ASME Section III, NB/NC/ND-2121 (a) requirements for compliance with the specifications for material given in ASME Section II, Part D (previously Section III, Division 1, Appendix I), Subpart 1, Tables 1 A, 1 B, 2A or 2B, as applicable, on the basis that compliance with code results in hardship without a compensating increase in quality.

Exelon desires to use nonstandard, proprietary, welded or nonwelded pipe fittings in applications requiring compliance with ASME Section III, without having to comply with the limitations on material specifications and grades mandated by Section III. Compliance with the ASME Code results in additional critical path time, cost, and radiation exposure that can be avoided through use of the code case.

Code Case N-879 ("Use of Micro-Alloyed Carbon Steel Bar in Patented Mechanical Joints and Fittings, Classes 1, 2, and 3 Section III, Division 1,") permits use of a micro-alloyed steel composition similar to that of ASME SA-675 and ASTM A 576 Grade 1524, with additions of vanadium and nitrogen, to enhance the strength needed to ensure a high-strength, leak• tight mechanical joint. The additional strength ensures that the deformation produced during installation of the fittings occurs in the pipe material, rather than in the fitting material. It is this deformation that produces the stresses necessary for the joint structural and leak-tight integrity.

This Code Case will expand Exelon's ability to use these proprietary fittings in safety• related piping, by including coverage for ASME Section III, Class 1, 2, and 3 systems NPS 2 or smaller. These provisions may also be used for installation of these fittings in B31.7 Class 1, 2, and 3 piping NPS 2 or smaller.

These fittings are already permitted to be used in safety-related piping constructed in accordance with ASME B31.1. These fittings are also already permitted to be used in compression-type fittings in ASME Section III, Class 1, 2, and 3 instrument lines, up to NPS 1, in accordance with NB/NC/N0-2121 (f) in the Winter 1972 Addenda through Winter 1973 Addenda, and NB/NC/N0-2121(d) through the 2017 Edition.

Most piping fabrication and installation joints have been traditionally fabricated by welding. Installation of pipe and piping subassemblies by mechanical means can save significant amounts of time, money, critical path time, and radiation exposure to plant personnel and installation and examination contractors. In systems containing radioactive materials, or in systems near irradiated components, personnel can be subjected to significant amounts of radiation during preparation for welding, welding, and nondestructive examination (NOE) of welds. Most of this exposure can be eliminated by use of mechanical connections. The amount of time to which mechanical installation personnel are exposed is a fraction of the time to which a welder or a nondestructive examiner would be exposed. Without installation welds, there is no associated installation NOE.

5. Proposed Alternative and Basis for Use:

Exelon proposes to implement the requirements of Code Case N-879 for procurement of nonstandard, proprietary welded and nonwelded pipe fittings NPS 2 or smaller.

ASME Section XI requires the fittings to be designed and manufactured in accordance with the original Construction Code, which, for these applications, is ASME Section III. These fittings are typically designed in accordance with ASME Section III, NB-3671.7, "Sleeve Coupled and Other Patented Joints," using the option of prototype testing. Alternatively, NC/N0-3671.7 may be
used for Class 2 or 3 fittings, as applicable.

Reconciliation and use of editions and addenda of ASME Section III will be in accordance with ASME Section XI, IWA-4220, and only editions and addenda of ASME Section Ill that have been accepted by 1 O CFR 50.55a may be used. The Code of Record for the specific 10-year ISI interval at each nuclear unit as identified under Section 2 above, will be used when applying the requirements of Section XI, unless specific regulatory relief to use of other editions or addenda is approved.

All other ASME Section XI requirements for which relief was not specifically requested and authorized by the NRG Staff will remain applicable, including third party review by the Authorized Nuclear lnservice Inspector.

Without the use of this Case in some situations, outage times could be increased, and plant and contractor personnel will receive significantly higher radiation doses, due to longer exposure times in the vicinity of the piping joint installation.

Based on the above, use of Code Case N-879 applies when compliance with the ASME Section III requirement for compliance with the specifications for material given in ASME Section II, Part O (previously Section III, Division 1, Appendix I), Subpart 1, Tables 1A, 1 B, 2A or 2B, as applicable, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Code Case N-879 was approved by the ASME Board on Nuclear Codes and Standards on May 10, 2017. It has not yet been incorporated into NRC Regulatory Guide 1.147, "lnservice Inspection Code Case Acceptability, ASME Section XI, Division 1," and thus is not available for application at nuclear power plants without specific NRC approval. Therefore, Exelon requests use of the alternative material requirements described in this Code Case via this relief request.

6. Duration of Proposed Alternative:

The proposed alternative is for use of the Case for the remainder of each plant's 10-year inspection interval as specified in Section 2 and for the remainder of the plant's life.

7. Precedent:

None

Exelon Generation

Exelon Generation

200 Exelon Way
Kennett Square. PA 19348

www.exeloncorp com

10CFR50.55A

RS-18-070
NMP1L3224
JAFP-18-0053
TMI-18-070

May 30, 2018

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-37 andNPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

James A. FitzPatrick Nuclear Power Plant
Renewed Facility Operating License No. DPR-59
NRC Docket No. 50-333

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Nine Mile Point Nuclear Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-63 and NPF-69
NRC Docket Nos. 50-220 and 50-410

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

R.E. Ginna Nuclear Power Plant
Renewed Facility Operating License No. DPR-18
NRC Docket No. 50-244

Three Mile Island Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Proposed Alternative to Utilize Coe Cases N-878 and N-880

In accordance with 10 CFR 50.55a(z)(2), Exelon Generation Company, LLC (Exelon) is requesting a proposed alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for lnservice Inspection of Nuclear Power Plant Components," on the basis that compliance with the code results in hardship without a compensating increase in quality. Specifically, this proposed alternative concerns the use of Code Case N-878 ("Alternative to QA Program Requirements of IWA-4142 Section XI, Division 1") and N-880 ("Alternative to Procurement Requirements of IWA-4143 for Small Nonstandard Welded Fittings Section XI, Division 1 "). These Code Cases address the procurement of material from a material supplier that does not possess ASME accreditation as a Quality System Certificate Holder or an NPT Certificate Holder. A separate relief request being submitted under a separate cover, requests the use of Code Case N-879 which allows the use of material that does not comply with the limitations on material specifications and grades mandated by ASME, Section III.

There are no regulatory commitments contained in this letter. Exelon requests your review and approval of this fleet request by May 30, 2019.

If you have any questions, please contact Tom Loomis (610) 765-5510.

Respectfully,

James Barstow
Director - Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachment: Proposed Alternative to Utilize Code Cases N-878 and N-880

cc: Regional Administrator - NRC Region I
Regional Administrator NRC Region Ill
NRC Senior Resident Inspector - Braidwood Station
NRC Senior Resident Inspector - Byron Station
NRC Senior Resident Inspector - Calvert Cliffs Nuclear Power Plant
NRC Senior Resident Inspector - Clinton Power Station
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - James A. FitzPatrick Nuclear Power Plant
NRC Senior Resident Inspector - LaSalle County Station
NRC Senior Resident Inspector - Limerick Generating Station
NRC Senior Resident Inspector - Nine Mile Point Nuclear Station
NRC Senior Resident Inspector - Peach Bottom Atomic Power Station
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
NRC Senior Resident Inspector - R.E. Ginna Nuclear Power Plant
NRC Senior Resident Inspector - Three Mile Island Nuclear Station, Unit 1
NRC Project Manager - Braidwood Station
NRC Project Manager - Byron Station
NRC Project Manager - Calvert Cliffs Nuclear Power Plant
NRC Project Manager - Clinton Power Station
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NRC Project Manager - Limerick Generating Station
NRC Project Manager - Nine Mile Point Nuclear Station
NRC Project Manager - Peach Bottom Atomic Power Station
NRC Project Manager - Quad Cities Nuclear Power Station
NRC Project Manager - R.E. Ginna Nuclear Power Plant
NRC Project Manager - Three Mile Island Nuclear Station, Unit 1

Attachment
Proposed Alternative to Utilize Code Cases N-878 and N-880

1. ASME Code Component(s) Affected:

All ASME Class 1, 2, and 3 carbon steel and stainless-steel piping systems.

2. Applicable Code Edition and Addenda:

PLANT	INTERVAL	EDITION	START	END
Braidwood Station, Units 1 and 2	Fourth	2013 Edition	August 29, 2018 October 17, 2018	July 28, 2028 October 16, 2028
Byron Station, Units 1 and 2	Fourth	2007 Edition, through 2008 Addenda	July 16, 2016	January 15, 2026
Calvert Cliffs Nuclear Power Plant, Units 1 and 2	Fourth	2004 Edition	October 10, 2009	June 30, 2019
Clinton Power Station, Unit 1	Third	2004 Edition	July 1, 2010	June 30, 2020
Dresden Nuclear Power Station, Units 2 and 3	Fifth	2007 Edition, through 2008 Addenda	January 20, 2013	January 19, 2023
James A. FitzPatrick Nuclear Power Plant	Fifth	2007 Edition, through 2008 Addenda	August 1, 2017	June 15, 2027
LaSalle County Stations, Units 1 and 2	Fourth	2007 Edition, through 2008 Addenda	October 1, 2017	September 30, 2027
Limerick Generating Station, Units 1 and 2	Fourth	2007 Edition, through 2008 Addenda	February 1, 2017	January 31, 2027
Nine Mile Point Nuclear Station, Unit 1	Fourth	2004 Edition	August 23, 2009	August 22, 2019

PLANT	INTERVAL	EDITION	START	END
Nine Mile Point Nuclear Station, Unit 2	Third	2004 Edition	April 5, 2008	August 22, 2018
Nine Mile Point Nuclear Station, Unit 2	Fourth	2013 Edition	August 23, 2018	August 22, 2028
Peach Bottom Atomic Power Station, Units 2 and 3	Fourth	2001 Edition, through 2003 Addenda	November 5, 2008	December 31, 2018
Peach Bottom Atomic Power Station, Units 2 and 3	Fifth	2013 Edition	January 1, 2019	December 31, 2028
Quad Cities Nuclear Power Station, Units 1 and 2	Fifth	2007 Edition, through 2008 Addenda	April 2, 2013	April 1, 2023
R.E. Ginna Nuclear Power Plant	Fifth	2004 Edition	January 1, 2010	December 31, 2019
Three Mile Island Nuclear Station, Unit 1	Fourth	2004 Edition	April 20, 2011	April 19, 2022

3. Applicable Code Requirements:

Code Case N-878 ("Alternative to QA Program Requirements of IWA-4142 Section XI, Division 1 ")

ASME Code, Section XI, IWA-4142 of the 2001 Edition with the 2003 Addenda, the 2004 Edition, the 2007 Edition with 2008 Addenda, and the 2013 Edition, provide requirements for procurement of materials to be used in repair/replacement activities.

Code Case N-880 ("Alternative to Procurement Requirements of IWA-4143 for Small Nonstandard Welded Fittings Section XI, Division 1")

ASME Code, Section XI, IWA-4142 and IWA-4143, of the 2001 Edition with the 2003 Addenda, the 2004 Edition, the 2007 Edition with 2008 Addenda, and the 2013 Edition, provide requirements for procurement of materials by the Owner, and fabrication of non-Code-stamped parts by the Owner, when the Construction Code is Section III, to be used in repair/replacement activities.

4. Reason for Request:

N-878

In accordance with 10 CFR 50.55a(z)(2), Exelon Generation Company, LLC (Exelon) is requesting proposed alternatives from the ASME Section XI, IWA-4200 requirements for compliance with Section III, NA-3700 or NCA-3800, as applicable, for procurement of nonstandard, nonwelded, proprietary pipe fittings larger than NPS 1 or Reactor Coolant System (RCS) makeup capacity, supplied as material, installed in Section XI repair/replacement activities in applications where the Construction Code is ASME Section III Winter 1973 addenda or later. Section XI, IWA-4142.1 of the 2007 Edition or later, specifies alternative procurement requirements for the Owner, but these alternatives may not be used by a non-ASME-accredited contracted Repair/Replacement Organization. Section XI permits use of these fittings in ASME B31.1, B31.7, or pre-Winter 1973 applications. Compliance with the ASME Code results in additional critical path time, cost, and radiation exposure that can be avoided through use of the Case.

N-880

In accordance with 1 O CFR 50.55a(z)(2), Exelon is requesting proposed alternatives from the ASME Section XI, IWA-4200 requirements for compliance with Section Ill, NA-8000 or NCA-8000, as applicable, for fabrication of nonstandard, proprietary welded pipe fittings larger than NPS 1 or RCS makeup capacity up to NPS 2, installed in Section XI repair/replacement activities in applications where the Construction Code is ASME Section III 1971 edition or later. Section XI, IWA-4143 permits fabrication of welded fittings at the Owner's facilities, but does not permit such fabrication to be performed in a facility owned by a Repair/Replacement Organization or other contractor or supplier. Compliance with the ASME Code results in additional critical path time, cost, and radiation exposure that can be avoided through use of the code case.

Both Cases

Exelon is requesting to use nonstandard, proprietary, welded or nonwelded pipe fittings in applications requiring compliance with ASME Section III, without having to comply with the administrative requirements imposed by ASME Section XI, IWA-4142, IWA-4143, and IWA-4200.

Nonstandard, proprietary welded or nonwelded pipe fittings can be proven, by testing, to comply with Section III design requirements. Exelon has a supplier of such fittings that does not possess ASME accreditation as a Quality System Certificate Holder or an NPT Certificate Holder.

Exelon is currently permitted by ASME Section XI to install the following:

1. Welded or nonwelded fittings produced by a non-ASME-accredited supplier in safety-related applications in ASME B31.1 or B31.7 Class I, II, or Ill piping systems (IWA-4221).

2. Welded or nonwelded fittings produced by a non-ASME accredited supplier in Class 1 systems no larger than NPS 1 and no larger than RCS makeup capacity (IWA-4131).

3. Welded or nonwelded fittings produced by this supplier in Class 2 or 3 systems NPS 1 or smaller (IWA-4131).

4. Nonwelded fittings produced by a non-ASME accredited supplier in ASME Section III Class 1, 2, and 3 piping systems as permitted by the reference code year.

5. Nonwelded fittings NPS 2 and smaller produced by this supplier in ASME Section III Class 1, 2, and 3 piping systems, provided Exelon verifies material conformance with the reference code year.

6. Nonwelded fittings larger than NPS 2 produced by a non-ASME-accredited supplier in ASME Section III Class 1, 2, and 3 piping systems, with additional material testing by Exelon as permitted by the reference code year.

7. Welded fittings fabricated by a non-ASME-accredited supplier in Class 2 and 3 piping systems in plants with construction permits issued before the NRG made Section III compliance and Code Symbol Stamping of Class 2 and 3 systems mandatory in 10 CFR 50.55a on May 14, 1984 (49 CFR 9711) (IWA-4221 and 10 CFR 50.55a).

Exelon is not currently permitted by ASME Section XI to install the following:

1. Nonwelded fittings larger than NPS 1, or RCS makeup capacity, fabricated by a non-ASME-accredited supplier, and purchased by a contractor without ASME accreditation without additional material testing by Exelon, in Section Ill, Class 1, 2, and 3 piping systems certified to the Section III Winter 1973 Addenda or later.

2. Welded fittings larger than NPS 1, or RCS makeup capacity, fabricated by a non- ASME accredited supplier in Section III, Class 1, 2, and 3 piping systems certified to the Section III 1971 Edition or later.

These two Cases will expand Exelon's ability to use these proprietary fittings in sizes larger than NPS 1, or RCS makeup capacity, in Section Ill, Class 1, 2, and 3 systems.

Most piping fabrication and installation joints have been traditionally fabricated by welding. Installation of pipe and piping subassemblies by mechanical means can save significant amounts of time, money, critical path time, and radiation exposure to plant personnel and installation and examination contractors. In systems containing radioactive materials, or in systems near irradiated components, personnel can be subjected to significant amounts of radiation during preparation for welding, welding, and nondestructive examination (NOE) of welds. Most of this exposure can be eliminated by use of mechanical connections. The amount of time to which mechanical installation personnel are exposed is a fraction of the time to which a welder or a nondestructive examiner would be exposed. Without installation welds, there is no associated installation NOE.

5. Proposed Alternative and Basis for Use:

Exelon proposes to implement the requirements of ASME Code Cases N-878 for procurement of nonstandard, nonwelded, proprietary pipe fittings larger than NPS 1, or RCS makeup capacity, supplied as material, and N-880 for procurement of nonstandard, proprietary welded pipe fittings larger than NPS 1, or RCS makeup capacity, up to NPS 2.

ASME Section XI requires the fittings to be designed in accordance with the original Construction Code, which, for these applications, is ASME Section III. These fittings are typically designed in accordance with ASME Section III, NB-3671.7, "Sleeve Coupled and Other Patented Joints," using the option of prototype testing. Alternatively, NC/N0-3671.7 may be used for Class 2 or 3 fittings, as applicable.

Reconciliation and use of editions and addenda of ASME Section III will be in accordance with ASME Section XI, IWA-4220, and only editions and addenda of ASME Section Ill that have been accepted by 10 CFR 50.55a may be used. The Code of Record for the specific 10-year ISI interval at each nuclear unit as identified under Section 2 above, will be used when applying the various IWA paragraphs of Section XI, unless specific regulatory relief to use other editions or addenda is approved.

All other ASME Section XI requirements for which relief was not specifically requested and authorized by the NRC Staff will remain applicable, including third party review by the Authorized Nuclear lnservice Inspector.

Without the use of these Code Cases in some situations, outage times could be increased, and plant and contractor personnel will receive significantly higher radiation doses, due to longer exposure times in the vicinity of the piping joint installation.

Based on the above, use of Code Cases N-878 and N-880 apply when compliance with the ASME Section III administrative requirement for possession of a Quality System Certificate (N-878) or NPT Certificate of Authorization (N-880) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Code Cases N-878 and N-880 were approved by the ASME Board on Nuclear Codes and Standards on April 18, 2017 and July 25, 2017, respectively. They have not yet been incorporated into NRC Regulatory Guide 1.147, "lnservice Inspection Code Case Acceptability, ASME Section XI, Division 1," and thus are not available for application at nuclear power plants without specific NRC approval. Therefore, Exelon requests use of the alternative procurement requirements described in these Cases via this relief request.

6. Duration of Proposed Alternative:

The proposed alternative is for use of the Cases for the remainder of each plant's to-year inspection interval as specified in Section 2 and for the remainder of the plant's life.

7. Precedent:

None

9.0 REFERENCES

The references used to develop this Inservice Inspection Program Plan include:

9.1 JAF Fourth Interval ISI Program Plan, SEP-ISI-007

9.2 ASME Section XI, 2007 Edition, 2008 Addenda

9.3 *BWRVIP-130: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines – 2004 Revision,* EPRI, Palo Alto, CA: 2004. 1008192.

9.4 BWR Vessel and Internals Project, "Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)"

9.5 NRC Safety Evaluation, dated 9/15/2000 approving modifications to Generic Letter 88-01 Inspection Schedules

9.6 NRC Final Safety Evaluation of the "BWRVIP Vessel and Internals Project, BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75), "EPRI Report TR -113932, October 1999 (TACNO. MA5012), dated 5/14/2002

9.7 BWRVIP-61 "BWR Vessel and Internals Project, Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants

9.8 *BWRVIP-75-A: BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules,* EPRI, Palo Alto, CA: 2005. 1012621.

9.9 BWRVIP letter 2006-491 from William A. Eaton (BWRVIP Chairman) to Document Control Desk (NRC), "Implementation of Inspection Relief for Hydrogen Water Chemistry and Noble Metal Chemical Application (BWRVIP-62-A)," dated November 15, 2006

9.10 BWRVIP letter 2007-135 from Randy Stark (EPRI) to All BWRVIP Members, "NRC Acknowledgement of BWRVIP Letter on Implementation of Inspection Relief for Hydrogen Water Chemistry and Noble Metal Chemical Application"

9.11 *BWRVIP-62-A: BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection.* EPRI, Palo Alto, CA: 2010. 1021006.

9.12 James A. FitzPatrick Nuclear Power Plant Code Case N-716-1 Application

9.13 JAF Initial Fifth Interval ISI Program Plan, ISI-JAF-LPT5-PLAN

9.14 JAF FSAR

9.15 JAF TRM