



Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-18-123

December 27, 2018

10 CFR 50.55a

ATTN: Document Control Desk  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Unit 1  
Renewed Facility Operating License No. DPR-33  
NRC Docket No. 50-259

**Subject: Re-Submittal of Browns Ferry Nuclear Plant Unit 1, American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, Inservice Inspection (ISI) Program and Augmented Program, Second Ten Year Interval Request for Relief 1-ISI-27**

- References:
1. TVA letter to NRC, CNL-15-125, "Browns Ferry Nuclear Plant (BFN) Unit 1, American Society of Mechanical Engineers (ASME) Section XI, Inservice Inspection (ISI) and Augmented Program, Second Ten Year Interval Request For Relief 1-ISI-27," dated June 26, 2015 (ML15181A448)
  2. TVA letter to NRC, CNL-15-215, "Response to NRC Request for Additional Information Regarding Browns Ferry Nuclear Plant Unit 1, American Society of Mechanical Engineers Section XI, Inservice Inspection and Augmented Program, Second Ten Year Interval Request For Relief 1-ISI-27 (CAC No. MF6401)," dated October 27, 2015 (ML15300A472)
  3. TVA letter to NRC, CNL-15-236, "Revised Response to NRC Request for Additional Information Regarding Browns Ferry Nuclear Plant, Unit 1, American Society of Mechanical Engineers Section XI, Inservice Inspection and Augmented Program, Second Ten Year Interval Request For Relief 1-ISI-27 (CAC No. MF6401)," dated November 18, 2015 (ML15338A221)
  4. NRC letter to TVA, "Browns Ferry Nuclear Plant, Unit 1 - Alternative Relief Request 1-ISI-27 for Relief from the Reactor Vessel Circumferential Weld Examination Requirements of the ASME Code (CAC No. MF6401)," dated February 17, 2016 (ML16020A115)

In References 1 through 3, Tennessee Valley Authority (TVA) submitted relief request 1-ISI-27, and associated responses to Request for Additional Information (RAI), for the Browns Ferry Nuclear Plant (BFN) Unit 1 second ten-year inspection (ISI) interval that ended on June 1, 2017. This relief request proposed an alternative in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1) for certain reactor vessel circumferential weld examinations currently required by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for the period of extended operation ending December 20, 2033. Relief Request 1-ISI-27 was approved by the Nuclear Regulatory Commission (NRC) in Reference 4.

Subsequent to the NRC approval of relief request 1-ISI-27, TVA determined that there was an error in the revised Table 3 that was submitted to NRC in Reference 3. Specifically, the revised Table 3 did not use the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) chemistry factor (CF) data (BWRVIP-135, Revision 3, "BWR Vessel and Internals Project") for the limiting unit circumferential (girth) reactor pressure vessel (RPV) weld (Heat 406L44). Use of the BWRVIP ISP data evaluation results in the BFN Unit 1 mean Reference Temperature for Nil Ductility Transition ( $RT_{NDT}$ ) exceeding the NRC acceptance criterion of 129.4°F at the end of the period of extended operation [(38 effective full power years (EFPY))]. This error has been entered into the TVA corrective action program (CAP). As part of the corrective action for this issue, TVA has reviewed the information in Table 3 in comparison to BWRVIP-135, Revision 3, which has resulted in updated values for CF, Delta  $RT_{NDT}$  Without Margin (°F), Initial  $RT_{NDT}$  (°F), and Mean  $RT_{NDT}$  (°F). The revised Table 3 is provided in the enclosure to this letter.

Because the Table 3 in the Relief Request, as approved by the NRC in Reference 4, did not utilize ISP data, TVA is re-submitting this relief request for NRC approval in accordance with 10 CFR 50.55a(z)(1.) The enclosure to this letter provides the justification as to why the proposed alternative provides an acceptable level of quality and safety. Table 1 of the enclosure contains specific information associated with each weld for which TVA is requesting relief from reactor vessel circumferential weld examinations. The enclosed relief request supersedes in its entirety the one previously submitted in References 1 through 3.

TVA requests approval of this relief request within one year from the date of this letter. There are no new regulatory commitments associated with this submittal. Please address any questions regarding this request to Michael A. Brown at 423-751-3275.

Respectfully,



E. K. Henderson  
Director, Nuclear Regulatory Affairs

Enclosure

cc: See Page 3

U.S. Nuclear Regulatory Commission  
CNL-18-123  
Page 3  
December 27, 2018

Enclosure:

Tennessee Valley Authority Browns Ferry Nuclear Plant (BFN) Unit 1  
American Society of Mechanical Engineers (ASME) Section XI, Inservice  
Inspection (ISI) and Augmented Program Second Ten Year Interval Request  
for Relief 1-ISI-27

cc (Enclosure):

NRC Regional Administrator - Region II  
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant  
NRC Project Manager - Browns Ferry Nuclear Plant

## Enclosure

### Browns Ferry Nuclear Plant (BFN) Unit 1 American Society of Mechanical Engineers (ASME) Section XI, Inservice Inspection (ISI) and Augmented Program Second Ten Year Interval Request for Relief 1-ISI-27

#### I. ASME Code Components Affected

Permanent relief from Reactor Vessel (RV) Circumferential Shell Weld Examinations is requested for the five welds listed in Table 1.

TABLE 1

Weld Number and Description	Category and Exam Method	Item Number	ASME Code Class
No. C-4-5, Vessel Shell to Shell Weld	B-A, Volumetric	B1.11	1
No. C-3-4, Vessel Shell to Shell Weld	B-A, Volumetric	B1.11	1
No. C-2-3, Vessel Shell to Shell Weld	B-A, Volumetric	B1.11	1
No. C-1-2, Vessel Shell to Shell Weld (Located in Belt-line Region)	B-A, Volumetric	B1.11	1
No. C-BH-1, Vessel Shell to Bottom Head Weld	B-A, Volumetric	B1.11	1

#### II. ASME Code Edition and Addenda

ASME Section XI, 2007 Edition through 2008 Addenda.

#### III. Applicable Code Requirement

ASME Section XI, 2007 Edition through 2008 Addenda, Table IWB-2500-1, Examination Category B-A, Item B1.11, requires a volumetric examination of the RV circumferential shell welds each interval.

#### IV. Reason for Request

The Tennessee Valley Authority (TVA) is requesting a proposed alternative in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1) on the basis that the proposed alternative provides an acceptable level of quality and safety. The proposed alternative would provide relief from RV circumferential weld examinations currently required by ASME Code for the period of extended operation.

Permanent relief from RV circumferential weld examinations was approved for the Second Ten-Year ISI Inspection Interval for BFN Unit 1 in the NRC letter dated May 31, 2005, for the remaining term of operation under the original operating license that expired on December 20, 2013.

## Enclosure

Relief from RV circumferential weld examinations for the period of extended operation was discussed in NUREG-1843, "Safety Evaluation Report Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3," dated April 2006, Section 4.2.6. The NRC reviewed the TVA Time-Limited Aging Analysis (TLAA) on RV circumferential weld examination relief, as summarized in License Renewal Application (LRA) Section 4.2.6, and determined that TVA appropriately explained that the conditional failure probabilities for the RV circumferential welds are bounded by the staff analysis in the Safety Evaluation Report (SER) for the Boiling Water Reactor Vessel and Internals Project (BWRVIP) BWRVIP-05 report, dated July 28, 1998, and that BFN will be using procedures and training to limit cold over-pressure events during the period of extended operation for BFN. The NRC concluded that the TVA LRA Section 4.2.6 on TLAA, and LRA Section A.3.1.6 for the BFN RV circumferential weld examination relief will meet the requirements of 10 CFR 54.21(c)(1)(ii) with one exception that was addressed by letter dated May 25, 2005, in which TVA provided the RV circumferential weld examination analysis for BFN Unit 1 in a revised version of Updated Final Safety Analysis Report (UFSAR) Supplement A.3.1.6.

### V. Proposed Alternative and Basis for Use

#### Background:

For the previous inservice inspection interval, the following information from NUREG-1843 (Section 4.2.6.2) was provided as the basis for use of the proposed alternative to perform only RV longitudinal shell weld examinations.

*"The technical basis for relief is discussed in the staff's final SER concerning the BWRVIP-05 report, which is enclosed in a July 28, 1998, letter from Mr. G. C. Laines (NRC) to Mr. C. Terry (BWRVIP Chairman). In this letter, the staff concluded that since the failure frequency for RV circumferential welds in BWR plants is significantly below the criterion specified in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," and below the core damage frequency of any BWR plant, the continued inspection would result in a negligible decrease in an already acceptably low value of RV failure. Therefore, elimination of the inservice inspection (ISI) for RV circumferential welds is justified. The staff's letter indicated that BWR applicants may request relief from ISI requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RV welds by demonstrating that (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the staff's July 28, 1998 evaluation, and (2) the applicants have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the frequency specified in the staff's SER. The letter indicated that the requirements for inspection of circumferential RV welds during an additional 20-year license renewal period would be reassessed, on a plant-specific basis, as part of any BWR LRA. Therefore, the applicant must request relief from inspection of circumferential welds during the license renewal period per 10 CFR 50.55a.*

*Section A.4.5 of the BWRVIP-74 report indicates that the staffs SER of the BWRVIP-05 report conservatively evaluated the BWR RVs to 64 EFPY, which is 10 EFPY greater than what is realistically expected for the end of the license renewal period. The staff used the mean RTNDT value for materials to evaluate failure probability of BWR circumferential welds at 32 and 64 EFPY in the staff SER dated July 28, 1998. The neutron fluence used in this evaluation was the neutron fluence at the clad-weld (inner) interface."*

## Enclosure

Because the NRC analysis discussed in the BWRVIP-74 report is a generic analysis, TVA submitted plant-specific information to demonstrate that the beltline materials meet the criteria specified in the report. On May 12, 2004, TVA submitted a relief request concerning the examination of the BFN Unit 1 RV circumferential welds for the original license period.

In Request for Additional Information (RAI) 4.2.6-1, dated December 1, 2004, the NRC requested the RV circumferential weld examination relief analyses for BFN Unit 1. As noted in Section 4.2.6 of NUREG-1843, on January 31, 2005, in response to RAI 4.2.6-1, TVA submitted the following relief analyses related to the BFN Unit 1 RV circumferential weld examination:

*“The staff evaluation of BWRVIP-05 utilized the favor code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the RPV shell weld failure probabilities. Three key assumptions of the PFM analysis were (1) the neutron fluence was the estimated end-of-license mean fluence, (2) the chemistry values were mean values based on vessel types, and (3) the potential for beyond design basis events (DBEs) was considered.”*

*“The following table provides a comparison of the BFN Unit 1 RV limiting circumferential weld parameters to those used in the NRC evaluation of BWRVIP-05 for the first two key assumptions. Data provided in this table was supplied from Tables 2.6.4 and 2.6.5 of the Final Safety Evaluation of the BWRVIP-05 Report (NRC letter from Gus C. Lainas to Carl Terry, Niagara Mohawk Power Company, BWRVIP Chairman), ‘Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report’ (TAC No. M93925), July 28, 1998.”*

**TABLE 2**

Effects of Irradiation on RPV Circumferential Weld Properties for BFN Unit 1

Group	Babcock and Wilcox (B&W) 64 EFPY	BFN Unit 1 54 EFPY
<i>Cu %</i>	<i>0.31</i>	<i>0.27</i>
<i>Ni %</i>	<i>0.59</i>	<i>0.6</i>
<i>Weld Chemistry Factor (CF)</i>	<i>196.7</i>	<i>184</i>
<i>Fluence at Clad/Weld Interface 10<sup>19</sup> neutrons (n)/centimeter<sup>2</sup> (cm<sup>2</sup>)</i>	<i>0.19</i>	<i>0.2</i>
<i>Delta RT<sub>NDT</sub> Without Margin (°F)</i>	<i>109.4</i>	<i>104</i>
<i>Initial RT<sub>NDT</sub> (°F)</i>	<i>20</i>	<i>20</i>
<i>Mean RT<sub>NDT</sub> (°F)</i>	<i>129.4</i>	<i>124</i>
<i>P (F/E) NRC</i>	<i>4.83 x 10<sup>-4</sup></i>	_____
<i>P (F/E) BWRVIP</i>	_____	_____

*“The fluence assumed for Unit 1 is very conservative based on an extended shutdown period from 1985 to a restart in 2007, which will result in less than 32 EFPY of vessel exposure through the end of the extended period of operation. However, TVA conservatively chose to use the higher exposure of 54 EFPY to simplify the basis for the Unit 1 vessel evaluations. As shown in the table, the Unit 1 unirradiated weld RT<sub>NDT</sub> is identical to the reference B&W plant unirradiated weld RT<sub>NDT</sub> used in the NRC analysis,*

## Enclosure

*and the Unit 1 fluence value is approximately equivalent to that used in the NRC analysis. However, because the Unit 1 chemistry factor is less than the reference B&W plant, the mean  $RT_{NDT}$  values for BFN Unit 1 at 54 EFPY are bounded by the 64 EFPY Mean  $RT_{NDT}$  assumed by the NRC in its analysis. Accordingly, Unit 1 is bounded by the conditional failure probability calculated by the Staff for the limiting B&W vessel. An extension of this relief for the 60-year period will be submitted to the NRC for approval prior to entering the period of extended operation.”*

The NRC verified the accuracy of the mean Reference Temperature for Nil Ductility Transition ( $RT_{NDT}$ ) for the limiting beltline circumferential weld on BFN Unit 1 and found it acceptable. In the NRC evaluation of the BWRVIP-05 report, a fluence of  $0.19 \times 10^{19}$  n/cm<sup>2</sup> for B&W RVs was used for 64 effective full power years (EFPY) and the corresponding delta  $RT_{NDT}$  value was 109.4 °F. The delta  $RT_{NDT}$  value for the limiting beltline weld metal of BFN Unit 1 was less than the limiting delta  $RT_{NDT}$  value in the NRC staff's evaluation of BWRVIP-05 report, which is conservative. Therefore, BFN's calculated mean  $RT_{NDT}$  value for the limiting beltline weld metal was acceptable and met the requirements specified in the NRC approved SER for the BWRVIP-05 report.

The SER for the BWRVIP-05 report provides a limiting conditional failure probability of  $4.83 \times 10^{-4}$  per reactor-year for a limiting plant-specific mean  $RT_{NDT}$  of 129.4 °F for B&W fabricated RVs. The Low Temperature Over-Pressure (LTOP) transient frequency is the frequency of the transient occurring, determined as  $1 \times 10^{-3}$  per reactor-year in the evaluation of BWRVIP-05 report. The conditional failure probability is the probability of failure, if the event were to occur. The vessel failure frequency is the product of conditional failure probability and LTOP frequency. Comparing the information in the Reactor Vessel Internal Database (RVID) with that submitted in the analysis, the NRC confirmed that the mean  $RT_{NDT}$  of the RV circumferential welds at BFN Unit 1 was projected to be 124°F at the end of the period of extended operation (54 EFPY). The chemistry factor (CF), delta  $RT_{NDT}$ , and mean  $RT_{NDT}$  were calculated consistent with the guidelines of Regulatory Guide (RG) 1.99, Revision 2. Because the calculated value of mean  $RT_{NDT}$  for the RV circumferential welds at BFN Unit 1 was lower than that for the limiting plant-specific case for B&W fabricated RVs, the vessel failure frequencies of the BFN Unit 1 RV circumferential welds was shown to be less than  $4.83 \times 10^{-7}$  per reactor-year.

The NRC found that BFN's evaluation for this TLAA was acceptable because the BFN Unit 1 54 EFPY conditional failure probabilities for the RV circumferential welds were bounded by the NRC analysis in the SER for the BWRVIP-05 report and because BFN will be using procedures and training to limit cold over-pressure events during the period of extended operation. The analysis satisfied the evaluation requirements of the NRC SER for the BWRVIP-05 report.

### **Discussion:**

TVA letter CNL-13-148, dated December 18, 2013, submitted a license amendment request (LAR) to revise Browns Ferry Nuclear Plant, Unit 1, Technical Specifications (TS) for Limiting Condition for Operation (LCO) 3.4.9, "RCS Pressure and Temperature (P/T) Limits" (ML13358A067). This submittal satisfied the requirements of NUREG-1843, "Safety Evaluation Report Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3," dated April 2006 (ML061030032), commitment 39 that required the development and submittal of revised P/T limit curves for NRC approval prior to the period of extended operation. These revised P/T limit curves were developed based on analyses projected to the end of the period of extended operation as required by 10 CFR 54.21(c)(1)(ii). On February 2, 2015, the NRC issued License Amendment Number 287 for BFN Unit 1 approving the use of the revised P/T limit curves (ML14325A501).

## Enclosure

Table 3 provides a comparison of the BFN Unit 1 RV limiting circumferential weld parameters to those used in the NRC evaluation of BWRVIP-05 using data from the revised BFN Unit 1 P/T curves from the December 18, 2013, LAR. The data in Table 3 for the B&W 64 EFPY is taken from Tables 2.6.4 and 2.6.5 of the SER for the BWRVIP-05 Report, dated July 28, 1998. Fluence values associated with the revised P/T curves for BFN Unit 1 38 EFPY were calculated using the General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation, NEDC-32983P-A, Revision 2.

<b>Table 3</b>		
<b>Comparison of the BFN Unit 1 RV Limiting Circumferential Weld Parameters to those Used In the NRC Evaluation of BWRVIP-05</b>		
<b>Group</b>	<b>B&amp;W 64 EFPY</b>	<b>BFN Unit 1 38 EFPY</b>
Cu %	0.31	0.27 <sup>1</sup>
Ni %	0.59	0.60 <sup>1</sup>
CF	196.7	280 <sup>1</sup>
Fluence at Clad/Weld Interface 10 <sup>19</sup> n/cm <sup>2</sup>	0.19	0.128 <sup>1</sup>
Delta RT <sub>NDT</sub> Without Margin (°F)	109.4	131 (see below)
Initial RT <sub>NDT</sub> (°F)	20	20 <sup>1</sup>
Mean RT <sub>NDT</sub> (°F)	129.4	151 (see below)
P (F/E) NRC	4.83 x 10 <sup>-4</sup>	-
P (F/E) BWRVIP	-	-

### Adjusted CF

The CF for the weld material was determined using the guidelines in BWRVIP-135, Revision 3, which suggests the more conservative surveillance program data be used, instead of the values from the RG 1.99 Rev. 2 tables, due to scatter in the surveillance data exceeding the credibility criteria. The revised CF of 280 was determined as noted in footnote 1 below.

<sup>1</sup> Pages 33 and 34 of NEDC-33445P (NEDO-3345), Revision 0, "Pressure and Temperature Limits Report (PTLR) Up to 25 and 38 Effective Full-Power Years," December 2013



## Enclosure

### Fluence at Clad/Weld Interface

The fluence factor was calculated using the following equation from RG 1.99, Revision 2:

Fluence Factor =  $f^{0.28 - 0.1 \log f}$  where  $f$  is the fluence in units of  $10^{19}$  n/cm<sup>2</sup>

Using a fluence of 0.128 in accordance with Table B-5 of NEDC-33445P, Revision 0:

Fluence Factor =  $0.128^{(0.28 - 0.1 \log 0.128)} = 0.468$

Delta  $RT_{NDT}$  = (Adjusted Chemistry Factor) \* (Fluence Factor)

Delta  $RT_{NDT}$  =  $280 * 0.468 = 131$

Mean  $RT_{NDT}$  = Initial  $RT_{NDT}$  + Delta  $RT_{NDT}$

Mean  $RT_{NDT}$  =  $131 + 20 = 151$

Using the data from BWRVIP-135 Revision 3 and the revised BFN Unit 1 P/T curves from the December 18, 2013, LAR, the BFN Unit 1 projected mean  $RT_{NDT}$  value is greater than the mean  $RT_{NDT}$  value from the NRC SER of BWRVIP-05, which means that BFN Unit 1 vessel is not bounded by this analysis. Therefore, TVA performed a plant specific analysis.

### **Plant Specific Evaluation:**

The plant specific analysis was performed using the methodology outlined in BWRVIP-05. A Monte Carlo simulation was performed to determine the beltline and axial circumferential weld failure probability using the software VIPER Version 1.2. This is the same methodology and software utilized in BWRVIP-05, Section 8.0, Probabilistic Fracture Mechanics. The evaluation results were compared to the NRC safety goal of  $1 \times 10^{-6}$  per year, as discussed in the NRC SER for BWRVIP-05. Inputs for the Monte Carlo simulation included BFN's Unit 1 surveillance chemistry data from BWRVIP-135, Revision 3.

Using the Monte Carlo simulation, the probability of failure (PoF) of the RV is defined as:

$$PoF = \frac{\text{Number of Failures}}{\text{Number of Simulations}}$$

The adjusted reference temperature (ART) at 25 EFPY and 38 EFPY was calculated using the following relation from Reg. Guide 1.99:

$$ART = \text{Initial } RT_{ndt} + \Delta RT_{ndt} + \text{Margin}$$

where  $\Delta RT_{ndt} = CF * FF$   
CF = chemistry factor  
FF = fluence factor

In this evaluation, the vessel failure is defined as when the applied stress intensity factor is equal to or larger than the material fracture toughness,  $K_{IC}$ , when the vessels experience an LTOP event per year. Based on the Final Safety Evaluation of the BWR Vessel and Internals

## Enclosure

Project Report, BWRVIP-05, the probability of an LTOP event is assumed to be  $1 \times 10^{-3}$  over the 42 years of plant life, resulting in a failure frequency of  $2.38 \times 10^{-5}$  per year.

One million vessel simulations were performed using the Monte Carlo probabilistic analysis techniques. The analysis showed that there were 13,659 brittle fracture failures of the RPV beltline circumferential weld as a result of an LTOP event, without inspection. This equates to a conditional failure probability of  $1.366 \times 10^{-2}$ .

In accordance with the TVA calculation that examined the Browns Ferry Unit 1 RPV circumferential weld relief with surveillance data from surveillance sample BFN 406L44, the resulting PoF for the case of no inspections of the RPV beltline circumferential welds is  $3.25 \times 10^{-7}$  per operating year. This value meets the NRC safety goal of  $1 \times 10^{-6}$  per vessel year for RPV failure frequency in BWRVIP-05 and the associated SER. The calculated PoF of  $3.25 \times 10^{-7}$  is also less than the regulatory limit of  $5 \times 10^{-6}$  events per year on an annual probability of developing a through-wall crack, as specified in NUREG-1874, which was published subsequent to BWRVIP-05.

### Conclusion:

The plant specific analysis shows that the BFN Unit 1 RPV PoF due to a LTOP event remains acceptable and the circumferential welds still qualify for reduced inspection through the period of extended operation (up to 38 EFPY), even with the elevated chemistry factor from surveillance sample BFN 406L44. Based on the fact that the PoF of the RPV beltline circumferential weld (i.e.,  $3.25 \times 10^{-7}$  per operating year) meets the NRC safety goal for RPV failure frequency in BWRVIP-05 (i.e.,  $1 \times 10^{-6}$  per year), TVA is requesting relief from the BFN Unit 1 reactor vessel circumferential weld examination of the ASME Code Section XI, Subarticle IWB-2500, Table IWB-2500-1, Examination Category B-A, Item Number B1.11 for RPV circumferential shell welds since the proposed alternative provides an acceptable level of quality and safety in accordance with 50.55a(z)(1).

### VI. Duration of Proposed Alternative

Relief is requested for examinations of RV circumferential welds for the period of the renewed license for BFN Unit 1 that expires December 20, 2033.

### VII. Precedents

Similar relief was approved for Oyster Creek Nuclear Generating Station (ML092520039) and Peach Bottom Atomic Power Station Units 2 and 3 (ML112770217).

### VIII. Attachment

Brown Ferry Unit 1 RPV shell weld location schematic drawing.

