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September 26, 2023 10 CFR 50.55a

RS-23-091

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

Byron Station, Unit 1
Renewed Facility Operating License No. NPF-37
NRC Docket No. 50-454

Subject: Relief Request I4R-25, Alternative Requirements for Reactor Pressure Vessel Inservice Inspection Intervals

In accordance with 10 CFR 50.55a, "Codes and standards," paragraph (z)(1), Constellation Energy Generation (CEG), LLC requests U.S. Nuclear Regulatory Commission (NRC) approval of the attached relief request associated with the fourth inservice inspection interval (ISI) for Byron Station (Byron), Unit 1. Relief is requested to allow for better project planning and work coordination which minimizes outage execution risk. As such, an alternative is requested to perform the volumetric examination of Byron Unit 1 Reactor Pressure Vessel welds, pertaining to Examination Category B-A and B-D, during the fourth ISI interval in the spring 2026 refueling outage. The details of the 10 CFR 50.55a request is enclosed.

CEG requests authorization of this request by January 2, 2024. There are no regulatory commitments contained within this letter.

Should you have any questions concerning this letter, please contact Brian Seawright at 779-231-6151.

Respectfully,

Lueshen, Kevin Digitally signed by Lueshen, Kevin
Date: 2023.09.26 08:10:03 -05'00'

Kevin Lueshen
Sr. Manager Licensing
Constellation Energy Generation, LLC

Attachment:

1. 10 CFR 50.55a Relief Request I4R-25, Revision 0

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Byron Station
Illinois Emergency Management Agency – Department of Nuclear Safety

ATTACHMENT 1

Byron Station
Unit 1

10 CFR 50.55a Relief Request I4R-25, Revision 0
Alternative Requirements for Reactor Pressure Vessel Inservice Inspection Intervals in Accordance with 10 CFR 50.55a(z)(1)

Request for Relief
Alternative Requirements for Reactor Pressure Vessel Inservice Inspection Intervals In Accordance with 10 CFR 50.55a(z)(1)

1.0 ASME CODE COMPONENTS AFFECTED:

The affected component is the Byron Station, Unit 1 reactor pressure vessel (RPV), specifically, the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI (Reference 1) examination categories and item numbers covering examinations of the RPV.

These affected Class 1 examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME B&PV Code, Section XI.

Examination Category	Item Number	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds

See Table 1 for a listing of the applicable examination areas of the Unit 1 RPV.

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The Fourth Interval Inservice Inspection (ISI) program is based on the ASME B&PV Code, Section XI, 2007 Edition through the 2008 Addenda (Reference 1).

The applicable edition for the ASME B&PV Code, Section XI for subsequent ISI intervals will be implemented in accordance with the requirements of 10 CFR 50.55a.

3.0 APPLICABLE CODE REQUIREMENT:

ASME Section XI, Paragraph IWB-2411 (Reference 1), "Inspection Program," requires volumetric examination of RPV welds identified in Table IWB-2500-1 once each inspection interval. The Byron Station, Unit 1 Fourth ISI Interval ends on July 15, 2026.

4.0 REASON FOR REQUEST:

As described under the Nuclear Regulatory Commission (NRC) Safety Evaluation (SE) for relief request I3R-23 (Precedent 1), Exelon Generation Company, LLC (EGC), now known as Constellation Energy Generation, LLC (CEG), proposed to defer the volumetric examinations of Byron Unit 1 RPV welds, pertaining to ASME B&PV Code Examination Category B-A and B-D, for the third ISI interval which ended on July 15, 2016. The third ISI interval was based on the ASME B&PV Code, Section XI, 2001 Edition through the 2003 Addenda. EGC proposed to perform these ASME B&PV Code required volumetric examinations during the fourth ISI interval for Byron Unit 1 in 2025.

However, CEG is planning to reschedule these examinations to the spring 2026 refueling outage to increase performance efficiencies with other major inspections and maintenance projects due to the longer refueling outage duration during that year. This allows for better project planning and work coordination which minimizes outage execution risk. This also eliminates an additional core barrel removal during the fall 2024 refueling outage (original planned date for the RPV welds examinations) which results in a reduction in industrial safety hazards and person-rem exposure.

Pursuant to 10 CFR 50.55a(z)(1), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety. An alternative is requested to perform the volumetric examination of Byron Unit 1 RPV welds, pertaining to Examination Category B-A and B-D, during the fourth ISI interval in the spring 2026 refueling outage. As the third ISI interval ended on July 15, 2016, the spring 2026 refueling outage is within the timeframe specified in Plant-Specific Information Item (3) under Section 2.4 of NRC SE for relief request I3R-23 (Precedent 1).

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

Constellation Energy Generation, LLC (CEG) proposes to perform the volumetric examinations of Byron Unit 1 RPV welds, pertaining to ASME B&PV Code Examination Category B-A and B-D, during the fourth ISI interval in the spring 2026 refueling outage. This proposed inspection date is within the timeframe specified in Plant-Specific Information Item (3) under Section 2.4 of NRC SE for relief request I3R-23 (Precedent 1).

The proposed inspection date is a deviation from the latest revised implementation plan, Pressurized Water Reactor Owners Group (PWROG) letter OG-10-238 (Reference 3), since the implementation plan reflects the next inspection being performed in 2025. The impact to the implementation plan in PWROG letter OG-10-238 (Reference 3) would decrease the number of inspections in 2025 from two to one and increase the number of inspections in 2026 from two to three. Based on Figures 3 and 4 of PWROG letter OG-10-238 (Reference 3), this proposed inspection schedule is considered to have a minor impact on the future inspection plan and the distribution of inspections over time.

In accordance with 10 CFR 50.55a(z)(1), an alternate inspection interval is requested on the basis that the current examination interval can be extended based on a negligible change in risk when compared to the risk criteria specified in Regulatory Guide 1.174 (Reference 4). The methodology used to conduct this analysis is based on that defined in WCAP-16168-NP-A, Revision 3 (Reference 5). This methodology focuses on risk assessments of materials within the beltline region of the RV wall. The results of the calculations for Byron Station, Unit 1 were compared to those obtained from the Westinghouse pilot plant evaluated in WCAP-16168-NP-A, Revision 3. Appendix A of WCAP-16168-NP-A, Revision 3 identifies the parameters to be compared. By demonstrating that the parameters for Byron Station, Unit 1 are bounded by the results of the Westinghouse pilot plant, the methodology qualifies Byron Station, Unit 1 for an ISI interval extension.

The following Table 1 lists the applicable examination areas addressed under this relief request of the Byron Station, Unit 1 RPV.

Table 1: Examination Category, Item Number, Component Description, and Component Identification of Applicable Examinations for Byron Unit 1

Exam Category	Item #	Component Description	Component Identification (RPV #/Exam Area)
Shell Components (In order from top to bottom of RPV)			
B-A	B1.30	Flange to Nozzle Belt (Upper Shell)	1RC-01-R/WR-7
B-A	B1.11	Nozzle Belt (Upper Shell) to Mid Shell	1RC-01-R/WR-34
B-A	B1.11	Mid Shell to Lower Shell	1RC-01-R/WR-18
B-A	B1.11	Lower Shell to Bottom Head Torus	1RC-01-R/WR-29
B-A	B1.21	Bottom Head Torus to Head Dome	1RC-01-R/WR-16
Nozzle Components			
B-D	B3.90	Outlet Nozzle @ 22°	1RC-01-R/RPVN-A
B-D	B3.90	Inlet Nozzle @ 67°	1RC-01-R/RPVN-B
B-D	B3.90	Inlet Nozzle @ 113°	1RC-01-R/RPVN-C
B-D	B3.90	Outlet Nozzle @ 158°	1RC-01-R/RPVN-D
B-D	B3.90	Outlet Nozzle @ 202°	1RC-01-R/RPVN-E
B-D	B3.90	Inlet Nozzle @ 247°	1RC-01-R/RPVN-F
B-D	B3.90	Inlet Nozzle @ 293°	1RC-01-R/RPVN-G
B-D	B3.90	Outlet Nozzle @ 338°	1RC-01-R/RPVN-H

Table 2 below lists the critical parameters investigated in WCAP-16168-NP-A, Revision 3 and compares the results of the Westinghouse pilot plant to those of Byron Station, Unit 1.

Table 2: Critical Parameters for the Application of Bounding Analysis for Byron Unit 1			
Parameter	**Pilot Plant Basis**	**Plant-Specific Basis**	**Additional Evaluation Required?**
Dominant Pressurized Thermal Shock (PTS) Transients in the NRG PTS Risk Study are Applicable	NRG PTS Risk Study (Reference 6)	PTS Generalization Study (Reference 7)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 5)	2.30E-14 Events per year (Calculated per Reference 5)	No
Frequency and Severity of Design Basis Transients	7 heatup/cooldown cycles per year (Reference 5)	Bounded by 7 heatup/cooldown cycles per year	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 5)	Single Layer	No

Table 3 below provides a summary of the latest RV inspection results for Byron Station, Unit 1 and evaluation of the recorded indications. This information confirms that satisfactory examinations have been performed on Byron Station, Unit 1 RV.

<table>
<tr><th colspan="2">Table 3: Additional Information Pertaining to RPV Inspection for Byron Unit 1</th></tr>
<tr><td>Inspection methodology:</td><td>The latest Unit 1 ISI was conducted in accordance with the ASME Code, Section XI and Section V, 1989 Edition, no Addenda. Examinations of Category B-A and B-D welds were performed to ASME Section XI Appendix VIII, 1995 Edition with the 1996 Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv and xvi).

Future inservice inspections will be performed to the applicable ASME Section XI Appendix VIII and 10 CFR 50.55a requirements.</td></tr>
<tr><td>Number of past inspections:</td><td>Two inservice inspections have been performed.</td></tr>
<tr><td>Number of indications found:</td><td>There were two indications identified in the beltline region circumferential weld during the most recent inservice inspection. These indications are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. Only one indication, contained in the nozzle shell forging material, is within the inner 1/10th or 1" of the RPV thickness. This indication is acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 8), since the flaw is less than the allowable number of flaws for each flaw size increment. A disposition of this flaw against the limits of the Alternate PTS Rule is shown in the table below.

<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in.)</th><th rowspan="2">Scaled Maximum number of forging flaws</th><th rowspan="2">Number of forging Flaws (Axial/Circ.)</th></tr>
<tr><th>TWEMIN</th><th>TWEMAX</th></tr>
<tr><td>0.075</td><td>0.375</td><td>76</td><td>1 (1/0)</td></tr>
<tr><td>0.125</td><td>0.375</td><td>30</td><td>1 (1/0)</td></tr>
<tr><td>0.175</td><td>0.375</td><td>8</td><td>1 (1/0)</td></tr>
</table>
</td></tr>
<tr><td rowspan="2">Proposed inspection schedule for balance of plant life:</td><td>The third inservice inspection for Unit 1 was scheduled for 2015, as the third ISI interval for Unit 1 ended on July 15, 2016. This inspection will instead be performed during the spring 2026 refueling outage, which is prior to the end date (July 15, 2026) of the fourth ISI interval for Unit 1.</td></tr>
<tr><td>The proposed inspection date is a deviation from the latest revised implementation plan, Pressurized Water Reactor Owners Group (PWROG) letter OG-10-238 (Reference 3), since the implementation plan reflects the next inspection being performed in 2025. The impact to the implementation plan in PWROG letter OG-10-238 (Reference 3) would decrease the number of inspections in 2025 from two to one and increase the number of inspections in 2026 from two to three. Based on Figures 3 and 4 of PWROG letter OG-10-238 (Reference 3), this proposed inspection schedule is considered to have a minor impact on the future inspection plan and the distribution of inspections over time.</td></tr>
</table>

Table 4 summarizes the inputs and outputs for the calculation of through-wall cracking frequency (TWCF) for Byron Station, Unit 1.

Table 4: Details of TWCF Calculation for Unit 1 at 57 Effective Full Power Years (EFPY)								
Inputs								
Reactor Coolant System Temperature, T_{RCS}[ºF]: N/A					**T_{wall} [inches]: 8.625**			
No.	**Region and Component Description**	**Material Heat No.**	**Cu[(a)] [wt%]**	**Ni[(a)] [wt%]**	**R.G. 1.99 Pos.**	**CF[(a)] [ºF]**	**RTNDT(u)[(a)] [ºF]**	**Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]**
1	**Nozzle Shell Forging**	**123J218**	**0.05**	**0.72**	**1.1**	**31.0**	**30**	**1.15**
2	**NS Forging to IS Forging Circ. Weld WF-501**	**442011**	**0.03**	**0.67**	**2.1**	**26.1**	**10**	**1.15**
3	**Intermediate Shell Forging**	**5P-5933**	**0.04**	**0.74**	**2.1**	**30.6**	**40**	**3.21**
4	**IS Forging to LS Forging Circ. Weld WF-336**	**442002**	**0.04**	**0.63**	**2.1**	**66.5**	**-30**	**3.08**
5	**Lower Shell Forging**	**5P-5951**	**0.04**	**0.64**	**1.1**	**26.0**	**10**	**3.21**

Outputs						
Methodology Used to Calculate ΔT_{30}: Regulatory Guide 1.99, Revision 2[(b)]						
	Controlling Material Region No. (From Above)	**RT_{MAX-XX} [ºF]**	**Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]**	**FF (Fluence Factor)**	**ΔT_{30} [ºF]**	**$TWCF_{95-XX}$**
Limiting Forging - FO	**3**	**79.98**	**3.21**	**1.307**	**39.98**	**9.19E-15**
Limiting Circumferential Weld - CW	**3**	**79.69**	**3.08**	**1.297**	**39.69**	**0.00E+00**
$TWCF_{95-TOTAL}(\alpha_{FO}TWCF_{95-FO} + \alpha_{CW}TWCF_{95-CW})$:						**2.30E-14**

Notes:
(a) Data obtained from Reference 9.
(b) Reference 10.

In summary, the proposed inspection date does not affect the validity of any technical bases described under relief request I3R-23 (Precedent 1). With this proposed alternative, Byron Unit 1 still satisfies the applicable Plant-Specific Information Items specified in the NRC SE for I3R-23 (Precedent 1), thereby providing an acceptable level of quality and safety.

6.0 <u>DURATION OF PROPOSED ALTERNATIVE:</u>

This request is applicable to the Byron Station, Unit 1 ISI program for the fourth inspection interval.

7.0 PRECEDENTS:

1. "Byron Station, Unit No. 1 - Relief from Requirements of the ASME Code to Extend the Reactor Vessel Inservice Inspection Interval (TAC No. MF3596)," dated December 10, 2014, Relief Request I3R-23 (ADAMS Accession Number ML14303A506, correction letter ML15022A416).

8.0 REFERENCES:

1. ASME Boiler and Pressure Vessel Code, Section XI and Section 2007 Edition, with 2008 Addenda, American Society of Mechanical Engineers, New York.

2. OG-06-356, "Plan for Plant-Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval.' MUHP 5097-99, Task 2059," October 31, 2006.

3. OG-10-238, "Revision to the Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval.' PA-MSC-0120," July 12, 2010.

4. NRG Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.

5. WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval," October 2011.

6. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," March 2010 (ADAMS Accession No. ML070860156).

7. NRG Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004 (ADAMS Accession No. ML042880482).

8. Code of Federal Regulations, 10 CFR Part 50.61a, "Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Washington D. C., Federal Register, Volume 75, No. 1, dated January 4, 2010 and No. 22 with corrections to part (g) dated February 3, 2010, March 8, 2010, and November 26, 2010.

9. WCAP-17606-NP, Revision 0, "Byron Station Units 1 and 2 Reactor Vessel Integrity Evaluation to Support License Renewal Time-Limited Aging Analysis," December 2012.

10. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.