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L-15-231

10 CFR 50.55a(z)

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

SUBJECT:
Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
Request to Extend Certain Reactor Vessel Inspections From 10 to 20 Years
(Requests 2-TYP-3-BA-01 and 2-TYP-3-BN-01)

In accordance with the provisions of 10 CFR 50.55a(z), FirstEnergy Nuclear Operating Company (FENOC) hereby requests Nuclear Regulatory Commission (NRC) approval of proposed alternatives to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME BPV Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," paragraph IWB-2412, "Inspection Program B," subsubarticle IWA-2430, "Inspection Intervals," and Table IWB-2500-1, "Examination Categories," at Beaver Valley Power Station, Unit No. 2.

ASME BPV Code, Section XI, Paragraph IWB-2412 requires, in part, volumetric examination of essentially 100 percent of reactor vessel pressure-retaining examination category B-A and B-D welds once each 10-year inservice inspection (ISI) interval. Subsubarticle IWA-2430 allows each inspection interval to be reduced or extended by as much as one year. Table IWB-2500-1 requires, in part, examination of certain reactor vessel welds and core support structure surfaces, once each ten year interval. The proposed alternatives would extend the ISI interval from 10 to 20 years for certain reactor vessel welds, and core support structure surfaces. A more detailed description of the proposed alternatives and supporting information are presented in the enclosures. FENOC requests approval of the proposed alternatives by December 31, 2016.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 315-6810.

Sincerely,

T.F. Stead for

Eric A. Larson

Enclosures:

1. Beaver Valley Power Station, Unit No. 2, 10 CFR 50.55a Request 2-TYP-3-BA-01, Revision 0
2. Beaver Valley Power Station, Unit No. 2, 10 CFR 50.55a Request 2-TYP-3-BN-01, Revision 0

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

--Alternative Provides Acceptable Level of Quality and Safety--

1. ASME Code Component(s) Affected

The affected component is the Beaver Valley Power Station, Unit No. 2 (BVPS-2) reactor vessel. Specifically, affected components include the pressure-retaining reactor vessel welds and full penetration reactor vessel nozzle welds listed below. The applicable American Society of Mechanical Engineers, Boiler and Pressure Vessel Code (ASME BPV Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," examination categories and item numbers are also listed. These examination categories and item numbers are from Subarticle IWB-2500 and Table IWB-2500-1 of ASME BPV Code, Section XI.

Examination Category	Item No.	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-A	B1.40	Head-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inside Radius Section

2. Applicable Code Edition and Addenda

ASME BPV Code, Section XI, 2001 Edition with the 2003 Addenda.

3. Applicable Code Requirement

ASME BPV Code Section XI, Paragraph IWB-2412, "Inspection Program B," requires volumetric examination of essentially 100 percent of the total number of reactor vessel pressure-retaining welds identified in Table IWB-2500-1, once each 10-year interval.

4. Reason for Request

An alternative is requested to the Paragraph IWB-2412 requirement that volumetric examination of essentially 100 percent of the total number of reactor vessel pressure-retaining Examination Category B-A and B-D welds be performed once each 10-year interval. Extension of the interval between examinations of Category B-A and B-D welds from 10 years to up to 20 years will result in a reduction in man-roentgen equivalent man (man-rem) exposure and examination costs.

5. Proposed Alternative and Basis for Use

FirstEnergy Nuclear Operating Company (FENOC) proposes to extend the third 10-year inservice inspection interval for BVPS-2 reactor vessel pressure-retaining Examination Category B-A welds and nozzle-to-vessel and nozzle inner radius section Examination Category B-D welds from August 28, 2018 to August 28, 2028. FENOC plans to perform the ASME BPV Code required volumetric examination of the BVPS-2 reactor vessel pressure-retaining Examination Category B-A welds and nozzle-to-vessel and nozzle inner radius section Examination Category B-D welds in 2027, since there is no refueling outage currently scheduled in 2028. The proposed inspection will be performed within the requested inservice inspection interval extension period, and within minus one refueling cycle (18 months) of the latest revised implementation plan date (2028) specified in Pressurized Water Reactor Owners Group (PWROG) Letter OG-10-238 (Reference 1).

In accordance with 10 CFR 50.55a(z)(1), an alternate inspection interval (extension of current interval) is requested based on a negligible change in risk by satisfying the risk criteria specified in Nuclear Regulatory Commission (NRC) Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 2).

The methodology used to conduct this analysis is based on Westinghouse topical report WCAP-16168-NP-A (the WCAP), "Risk Informed Extension of the Reactor Vessel In-service Inspection Interval" (Reference 3). This report focused on risk assessments of materials within the reactor vessel beltline region. The critical parameters for BVPS-2 were compared to those from the Westinghouse pilot plant evaluated in the WCAP. Appendix A of the WCAP identifies the parameters to be compared. Demonstrating that the parameters for BVPS-2 are bounded by the results of the Westinghouse pilot plant confirms that the WCAP is applicable and qualifies BVPS-2 for an inservice inspection interval extension.

Table 1 lists the critical parameters for application of the bounding analysis identified in Appendix A of the WCAP and compares the results of the Westinghouse pilot plant to those of BVPS-2.

Table 1: Critical Parameters for the Application of Bounding Analysis for BVPS-2			
Parameter	**Pilot Plant Basis**	**Plant-Specific Basis**	**Additional Evaluation Required?**
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are Applicable	NRC PTS Risk Study (Reference 5)	PTS Generalization Study (Reference 6)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events Per Year (Reference 3)	1.31E-11 Events Per Year (Calculated Per Reference 3)	No
Frequency and Severity of Design Basis Transients	7 Heatup/Cooldown Cycles Per Year (Reference 3)	Bounded by 7 Heatup/Cooldown Cycles Per Year	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 3)	Single Layer	No

The seven heatup/cooldowns per year assumed for the Westinghouse pilot plant study in the WCAP is used to demonstrate that the amount of fatigue crack growth considered in the pilot plant analyses is bounding for specific Westinghouse plants. The projected number of RCS heatup/cooldown cycles for 60 years of operation (109 each) are provided in Table 4.3-2 of the BVPS-2 License Renewal Application, (ADAMS Accession Number ML072470493), and the number of RCS heatup/cooldown design cycles (200 each) are listed in BVPS-2 UFSAR Table 3.9N-1. Over the 60 years of plant operation, the number of projected and design RCS heatup/cooldown cycles per year for BVPS-2 are less than the bounding seven heatup/cooldown cycles per year.

Tables 2 and 3 provide the results of the BVPS-2 surveillance capsule data statistical tests required by paragraph (f)(6) of the alternate pressurized thermal shock rule (Alternate PTS Rule), 10 CFR 50.61a (Reference 4). As shown, the surveillance results satisfy the criteria in the Alternate PTS Rule for all three deviation tests for each material. Therefore, use of the equations in the Alternate PTS Rule for calculation of ΔT_{30} is acceptable for BVPS-2.

Table 2: Surveillance Data Evaluation for Intermediate Shell Plate B9004-2

Capsule	Direction	Log of Fluence	Residual "r"	$(x - x_{avg})^2$	r* (r/sigma)
U	Longitudinal	18.79	-5.5	0.349	-0.30
V	Longitudinal	19.42	-5.2	0.002	-0.28
W	Longitudinal	19.56	-0.5	0.032	-0.03
X	Longitudinal	19.75	8.3	0.138	0.44
U	Transverse	18.79	-11.8	0.349	-0.64
V	Transverse	19.42	-15.1	0.002	-0.81
W	Transverse	19.56	-8.1	0.032	-0.44
X	Transverse	19.75	14.4	0.138	0.77

Mean Deviation Test	
Standard Deviation (sigma)	18.6
Mean Deviation	-3.0
Maximum Mean Residual	15.3
Pass/Fail?	Pass

Slope Deviation Test	
Slope (m)	15.62
Standard Error of Fit	8.57
Standard Error of Slope	8.41
T-Statistic	1.86
Critical T-Statistic	3.14
Pass/Fail?	Pass

Outlier Deviation Test	
Largest r*	0.77
Largest allowable r*	3.02
Pass/Fail?	Pass
Second largest r*	0.44
Second largest allowable r*	2.05
Pass/Fail?	Pass

Table 3: Surveillance Data Evaluation for Weld Wire Heat 83642

Capsule	Direction	Log of Fluence	Residual "r"	$(x - x_{avg})^2$	r* (r/sigma)
U	N/A	18.79	-28.0	0.349	-1.06
V	N/A	19.42	-34.4	0.002	-1.30
W	N/A	19.56	-63.1	0.032	-2.39
X	N/A	19.75	-62.1	0.138	-2.35

Mean Deviation Test	
Standard Deviation (sigma)	26.4
Mean Deviation	-46.9
Maximum Mean Residual	30.8
Pass/Fail?	Pass

Slope Deviation Test	
Slope (m)	-36.84
Standard Error of Fit	12.27
Standard Error of Slope	17.02
T-Statistic	-2.16
Critical T-Statistic	6.96
Pass/Fail?	Pass

Outlier Deviation Test	
Largest r*	-1.06
Largest allowable r*	2.81
Pass/Fail?	Pass
Second largest r*	-1.30
Second largest allowable r*	1.73
Pass/Fail?	Pass

Tables 4 and 5 provide additional information required by the NRC staff and included in Appendix A of the WCAP. Table 4 below provides a summary of the latest reactor vessel inspection for BVPS-2 and an evaluation of the recorded indications. This information confirms that acceptable results have been obtained for inspections performed on the BVPS-2 reactor vessel.

Table 4: Additional Information Pertaining to Reactor Vessel Inspection for BVPS-2	
Inspection methodology:	The most recent inservice inspection of the Category B-A and B-D welds was performed to ASME BPV Code Section XI, 1989 Edition with no Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv and xvi) (implementation of Appendix VIII). Future inservice inspections will be performed to ASME Code Section XI, Appendix VIII requirements.
Number of past inspections:	Two 10-year inservice inspections have been performed.

<table>
<tr><th colspan="2">Table 4: Additional Information Pertaining to Reactor Vessel Inspection for BVPS-2</th></tr>
<tr><td>Number of indications found:

in. - inches
MIN - Minimum
MAX - Maximum
Circ. - Circumferential</td><td>There were 52 indications identified in the beltline region during the most recent inservice inspection. These indications are acceptable per Table IWB-3510-1 of Section XI of the ASME BPV Code. Five of these indications are within the inner $1/10^{th}$ or 1-inch of the reactor vessel thickness and are acceptable per the requirements of the Alternate PTS Rule. A disposition of these five flaws against the limits of the Alternate PTS Rule is shown in the tables below. The following indications are located within the weld material of the reactor vessel beltline.
<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in.)</th><th rowspan="2">Scaled maximum number of weld flaws</th><th rowspan="2">Number of weld flaws (Axial/Circ.)</th></tr>
<tr><th>TWE_{MIN}</th><th>TWE_{MAX}</th></tr>
<tr><td>0.075</td><td>0.475</td><td>131</td><td>1 (1/0)</td></tr>
<tr><td>0.125</td><td>0.475</td><td>71</td><td>1 (1/0)</td></tr>
</table>
The following indications are located within the plate material of the reactor vessel beltline.
<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in.)</th><th rowspan="2">Scaled maximum number of plate flaws</th><th rowspan="2">Number of plate flaws (Axial/Circ.)</th></tr>
<tr><th>TWE_{MIN}</th><th>TWE_{MAX}</th></tr>
<tr><td>0.075</td><td>0.375</td><td>53</td><td>4 (4/0)</td></tr>
<tr><td>0.125</td><td>0.375</td><td>21</td><td>4 (4/0)</td></tr>
</table>
The maximum number of allowable flaws specified in the Alternate PTS Rule were scaled based on the total inspected weld length (780.7 inches) and plate area (6,485.4 square inches) within the beltline region to determine the scaled number of flaws above.</td></tr>
<tr><td>Proposed inspection schedule for balance of plant life:</td><td>The third inservice inspection is scheduled for 2018. It is proposed that this inspection be performed in 2027. This inspection date is within minus one refueling cycle (18 months) of the latest revised implementation plan date (2028) specified in PWROG letter OG-10-238.</td></tr>
</table>

Table 5 summarizes the inputs and outputs for the calculation of through-wall cracking frequency (TWCF). Alternate PTS Rule equations were used to calculate the ΔT_{30} values. These equations, based in part on NUREG-1874 (Reference 5), are considered an NRC approved methodology for the purpose of this request.

Table 5: Details of TWCF Calculation for BVPS-2 at 54 Effective Full Power Years (EFPY)								
Inputs								
Reactor Coolant System Temperature, T_C [°F]:				541.6	T_{wall} [inches]:			8.031
No.	Region and Component Description	Material Heat No.	Cu(1) [wt%]	Ni(1) [wt%]	P [wt%]	Mn [wt%]	$RT_{NDT(u)}$ [°F]	Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]
1	Intermediate Shell Plate B9004-1	C0544-1	0.065	0.55	0.010(2)	1.40(3)	60	5.18
2	Intermediate Shell Plate B9004-2	C0544-2	0.06	0.57	0.007(2)	1.45(3)	40	5.18
3	Lower Shell Plate B9005-1	C1408-2	0.08	0.58	0.009(2)	1.33(3)	28	5.21
4	Lower Shell Plate B9005-2	C1408-1	0.07	0.57	0.009(2)	1.34(3)	33	5.21
5	IS Long. Weld 101-124A	83642	0.046	0.086	0.019(4)	1.63(4)	-30	1.76
6	IS Long. Weld 101-124B	83642	0.046	0.086	0.019(4)	1.63(4)	-30	1.76
7	LS Long. Weld 101-142A	83642	0.046	0.086	0.019(4)	1.63(4)	-30	1.77
8	LS Long. Weld 101-142B	83642	0.046	0.086	0.019(4)	1.63(4)	-30	1.77
9	IS to LS Circ. Weld 101-171	83642	0.046	0.086	0.019(4)	1.63(4)	-30	5.18

Outputs					
Methodology Used to Calculate ΔT_{30}:		10 CFR 50.61a(5)			
	Controlling Material Region No. (From Above)	RT_{MAX-XX} [°R]	Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]	ΔT_{30} [°F]	$TWCF_{95-XX}$
Limiting Axial Weld - AW	1	582.3	1.76	62.7	0.00E+00
Limiting Plate - PL	1	613.1	5.18	93.4	5.25E-12
Circumferential Weld - CW	1	613.1	5.18	93.4	0.00E+00
$TWCF_{95-TOTAL}(\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW})$: (6)					1.31E-11

Notes: (1) Reference 7

(2) Reference 8

(3) Taken from plant-specific certified material test reports

(4) Conservative estimates per 10 CFR 50.61a (Reference 4)

(5) Reference 4

(6) The α terms used in the table above are determined as shown on page 36 of Reference 5

Section 3.4 of the NRC safety evaluation incorporated in the WCAP includes plant specific information requests 4, 5, and 6. These requests are applicable to licensees with B&W plants, reactor pressure vessels having forgings that are susceptible to underclad cracking and with RT_{MAX-FO} values exceeding 240 degrees Fahrenheit, and licensees seeking second or additional interval extensions. Information requests 4, 5, and 6 do not apply to FENOC request 2-TYP-3-BA-01, Revision 0 for BVPS-2.

6. Duration of Proposed Alternative

The proposed alternative would extend the duration of the third 10-year inservice inspection interval for BVPS-2 reactor vessel pressure-retaining Examination Category B-A welds and nozzle-to-vessel and nozzle inner radius section Examination Category B-D welds to August 28, 2028.

7. Precedent

Similar requests by the Callaway Plant, Unit No. 1 and Diablo Canyon Power Plant, Unit No. 1 have been recently approved to extend the interval between examinations of Category B-A and B-D reactor vessel welds from 10 years to up to 20 years. These requests provided information to address the risk-informed criteria set forth in the WCAP. NRC letters authorizing the alternatives are referenced below.

1. Callaway Plant, Unit No. 1
   Docket No. 50-483, February 10, 2015 NRC Letter
   ADAMS Accession Number ML15035A148
   In this letter the NRC staff authorized extending the third 10-year ISI interval from December 18, 2014 to December 18, 2024 for the subject components.
2. Diablo Power Plant, Unit No. 1
   Docket No. 50-275, June 19, 2015 NRC Letter
   ADAMS Accession Number ML15168A024
   In this letter the NRC staff authorized extending the third 10-year ISI interval from May 6, 2015 to May 6, 2025 for the subject components.

8. References

1. 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 (ADAMS Accession Number ML11153A033).

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

3. WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," October 2011 (ADAMS Accession Number ML113060207).

4. 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.

5. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," March 2010.

6. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004 (ADAMS Accession Number ML042880482).

7. WCAP-16527-NP Supplement 1, Revision 1, "Analysis of Capsule X from FirstEnergy Nuclear Operating Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," September 2011 (ADAMS Accession Number ML13151A060).

8. Nuclear Regulatory Commission (NRC) Reactor Vessel Integrity Database (RVID), Version 2.0.1, July 2000, <http://www.nrc.gov/reactors/operating/ops-experience/reactor-vessel-integrity/database-overview.html#download>.

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1. ASME Code Component(s) Affected

The affected component is the Beaver Valley Power Station, Unit No. 2 (BVPS-2) reactor vessel, specifically, the reactor vessel sub-components listed below. The applicable American Society of Mechanical Engineers, Boiler and Pressure Vessel Code (ASME BPV Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," examination categories and item numbers are also listed. The listed examination categories (hereafter referred to as category) and item numbers are from Subarticle IWB-2500 and Table IWB-2500-1 of ASME BPV Code, Section XI.

Examination Category	Item No.	Description
B-N-2	B13.50	Interior Attachments Within Beltline Region
B-N-3	B13.70	Core Support Structure

2. Applicable Code Edition and Addenda

ASME BPV Code, Section XI, 2001 Edition with the 2003 Addenda.

3. Applicable Code Requirement

Subsubarticle IWA-2430, "Inspection Intervals," subparagraph (d)(1) states that:

> Each inspection interval may be reduced or extended by as much as one year. Adjustments shall not cause successive intervals to be altered by more than one year from the original pattern of intervals. If an inspection interval is extended, neither the start and end dates nor the inservice inspection program for the successive interval need be revised.

Table IWB-2500-1, "Examination Categories," includes the following reactor pressure vessel visual examinations once each 10-year interval, among other examination requirements:

- Item number B13.50, accessible interior attachment welds within the beltline region (category B-N-2), and
- Item number B13.70, accessible core support structure surfaces (category B-N-3).

4. Reason for Request

The current BVPS-2 third 10-year inservice inspection interval began on August 29, 2008 and is scheduled to end on August 28, 2018. Request 2-TYP-3-BA-01 proposes an alternative to the ASME BPV Code requirements pursuant to 10 CFR 50.55a(z)(1) on the basis that the extended inspection interval (20-year interval ending August 28, 2028) provides an acceptable level of quality and safety. Request 2-TYP-3-BA-01 would permit future examination of the pressure-retaining reactor pressure vessel welds and full penetration reactor pressure vessel nozzle welds (category B-A and B-D reactor pressure vessel welds) to be performed during the maintenance and refueling outage currently scheduled in 2027. Request 2-TYP-3-BN-01 proposes to extend the third inspection interval for visual examination of the interior attachment welds within the reactor pressure vessel beltline region (category B-N-2), and visual examination of the accessible core support structure surfaces (category B-N-3) in order to allow deferral of the subject examinations to the same refueling outage as the category B-A and B-D reactor pressure vessel shell welds and nozzle welds described in request 2-TYP-3-BA-01.

During the ten-year inservice inspection of the BVPS-2 reactor pressure vessel shell, lower head, and nozzle welds performed in 2008, FENOC also performed visual examinations of the accessible reactor pressure vessel interior attachment welds and the accessible core support structure surfaces. Since the core support structure requires removal to facilitate examination of the reactor pressure vessel shell, lower head, and nozzle welds, visual examinations of the reactor pressure vessel interior attachment welds and core support structure surfaces have historically been performed during the same outage at the end of the inservice inspection interval.

Performing these related examinations during the same refueling outage results in significant savings in outage duration since the same equipment and personnel used for examination of the reactor pressure vessel shell, lower head, and nozzle welds from the reactor pressure vessel interior can implement the examinations of reactor pressure vessel interior attachment welds and core support structure surfaces. Additionally, removing the reactor pressure vessel internals once instead of twice during the proposed 20-year inspection interval to perform these related examinations would result in significant savings in radiation exposure.

5. Proposed Alternative and Basis for Use

FENOC proposes to extend the third 10-year inservice inspection interval for the category B-N-2 interior attachment welds within the reactor vessel beltline region and the category B-N-3 reactor vessel core support structure surfaces until August 28, 2028. The subject examinations would need to be performed before the end of the Fall 2018 refueling outage, pending approval of this proposed alternative. The proposed alternative inspection would enable the subject examinations to be performed during the refueling outage in 2027 with the risk-informed extension of the inservice inspection interval for category B-A reactor vessel pressure-retaining welds and category B-D nozzle-to-vessel and inner radius section welds.

The visual examinations of the reactor pressure vessel interior attachment welds and core support structure surfaces have been performed twice at BVPS-2. There were no relevant indications noted during the examinations. The examinations were last performed during the 2008 maintenance and refueling outage with acceptable results. Additionally, review of industry surveys indicate that these examinations have been performed many times by the industry without any significant findings relevant to the BVPS-2 reactor vessel design.

During the 2015 refueling outage FENOC performed the category B-N-1 visual examination. This examination included the reactor vessel interior areas that are made accessible for examination by the removal of components during normal refueling outages. This examination is required once each inspection period and will provide reasonable assurance of structural integrity.

This request would extend the duration of the third 10-year inspection interval for category B-N-2 and B-N-3, Item Numbers B13.50 and B13.70, visual examinations to 20-years. The proposed inspection interval extension is an alternative to subarticle IWA-2430 requirements regarding inspection interval length and Table IWB-2500-1 item numbers B13.50 and B13.70 requirements regarding examination frequency. The fact that no relevant indications were noted during the previous examinations, including the category B-N-1 visual examination during the 2015 refueling outage, provides reasonable assurance of structural integrity.

6. Duration of Proposed Alternative

The proposed alternative would extend the duration of the third 10-year inspection interval for category B-N-2 and B-N-3, Item Numbers B13.50 and B13.70, visual examinations to August 28, 2028.

7. Precedent

Southern California Edison submitted a similar request for San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, proposing that the visual examinations for Category B-N-2 and B-N-3 components be performed consistent with the proposed inspection interval for Category B-A and B-D volumetric examinations. By letter dated November 4, 2011 (ADAMS Accession Number ML112730074), the NRC staff authorized use of the alternative.