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

July 2, 2018

Mr. Richard D. Bologna
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Beaver Valley Power Station
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SUBJECT: BEAVER VALLEY POWER STATION, UNIT 1 – MODIFIED PRESSURIZED THERMAL SHOCK REFERENCE VALUES AND SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE (EPID NO. L-2017-LLL-0024)

Dear Mr. Bologna:

By letter dated October 6, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17284A195), as supplemented by letters dated April 6, 2018 (ADAMS Accession No. ML18099A123), FirstEnergy Nuclear Operating Company (licensee) submitted a request to revise the pressurized thermal shock (PTS) reference temperature (RTPTS) values for Beaver Valley Power Station, Unit No. 1 (BVPS-1), reactor vessel beltline and extended beltline region materials. This submittal also requested approval of a modified reactor vessel surveillance capsule withdrawal schedule pursuant to 10 CFR 50, Appendix H, Section III, paragraph 8.3.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the submittal and concludes that the proposed changes are acceptable and consistent with the intent and requirements of the applicable regulations and guidance found in Appendix H to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, as well as American Society for Testing and Materials Standard E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," dated July 1, 1982. The NRC staff's related safety evaluation (SE) is enclosed.

The NRC staff notes that its review of Enclosure C to the licensee's October 6, 2017, submittal, WCAP-18102-NP, Revision 0, BVPS-1 "Heatup and Cooldown Limit Curves for Normal Operation" was limited to Appendixes E and G of WCAP-18102-NP, which contain the PTS evaluation and revised surveillance capsule withdrawal schedule. The licensee did not request review of, nor did the staff review, the main body of WCAP-18102-NP which addresses the pressure-temperature (P-T) limits for BVPS-1.

The enclosed SE does not approve the P-T limits in WCAP-18102-NP, Revision 0, or in Revision 1 to the same report, which was submitted via the licensee's April 6, 2018, letter.

Sincerely,

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

Docket No. 50-334

Enclosure:
Safety Evaluation

cc: Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FIRST ENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

MODIFIED PRESSURIZED THERMAL SHOCK REFERENCE VALUES AND SURVEILLANCE

CAPSULE WITHDRAWAL SCHEDULE

DOCKET NO. 50-334 RENEWED OPERATING LICENSE NO. DPR-66

1.0 INTRODUCTION

By letter dated October 6, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17284A195), as supplemented by letters dated April 6, 2018 (ADAMS Accession No. ML18099A123), FirstEnergy Nuclear Operating Company (licensee or FENOC) submitted a request to revise the pressurized thermal shock (PTS) reference temperature (RTPTS) values for Beaver Valley Power Station, Unit No. 1 (BVPS-1), reactor vessel beltline and extended beltline region materials. This submittal also requested approval of a modified reactor vessel surveillance capsule withdrawal schedule pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix H, Section III, paragraph 8.3.

Enclosure A of the licensee's submittal included a description of the pressurized thermal shock evaluation, and results that supports the request for approval of RTPTS values. The licensee stated that the updated RTPTS values incorporate Capsule X fluence analysis results, sister plant surveillance capsule test results, and revised unirradiated nil-ductility reference temperature ($RTNDT_{(u)}$) values for each of the four reactor vessel beltline plate materials. The licensee's evaluation of the proposed changes to the reactor vessel surveillance capsule withdrawal schedule was provided in Enclosure B.

Appendix E to Enclosure C of the licensee's submittal, WCAP-18102-NP, Revision 0, Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," contains the PTS evaluation for BVPS-1 reactor vessel beltline and extended beltline region materials at the end of 50 effective full power years (EFPY). The licensee provided WCAP-18102-NP, Revision 1, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," (ADAMS Accession No. ML18099A125) in its April 6, 2018, response to the U.S. Nuclear Regulatory Commission (NRC) staff's request for additional information (RAI), which the licensee stated contains a revised PTS evaluation. Appendix G to WCAP-18102-NP, Revision 0, contains the proposed revised surveillance capsule withdrawal schedule for BVPS-1. The staff notes that its review was limited to Appendices E and G of WCAP-18102-NP, Revision 0 and Revision 1. The licensee did not request review of, nor did the NRC staff review the main body of WCAP-18102-NP, Revision 0, which addresses the pressure-temperature (P-T) limits for BVPS-1, or Revision 1 of the same report. The most recent pressure-temperature limits report (PTLR) for BVPS-1 was submitted to the NRC for information on October 4, 2017 (ADAMS Accession No. ML17277B091), and references WCAP-18102, Revision 0. As stated in Generic Letter (GL) 96-03, the NRC staff does not normally review and approve PTLR revisions via a safety evaluation (SE) unless there is a change in the methodology used to generate the P-T limits.

2.0 REGULATORY EVALUATION

Per 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," the NRC requires that for each pressurized-water nuclear power reactor for which an operating license has been issued under 10 CFR Part 50 or a combined license issued under 10 CFR Part 52, other than a nuclear power reactor facility for which the certification required under 10 CFR 50.82(a)(1) has been submitted, the licensee shall have projected values of RT_{PTS} or RT_{MAX-X}, accepted by the NRC, for each reactor vessel beltline material. For pressurized-water nuclear power reactors for which a construction permit was issued under this part before February 3, 2010, and whose reactor vessel (RV) was designed and fabricated to the 1998 Edition or earlier of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) (applicable to BVPS-1), the projected values must be in accordance with 10 CFR 50.61 or 10 CFR 50.61a.

The RT_{PTS} means the reference temperature, RT_{NDT}, evaluated for the end-of life (EOL) fluence for each of the vessel beltline materials, using the procedures of paragraph (c) of 10 CFR 50.61. The RT_{PTS} is calculated by summing the initial (unirradiated) RT_{NDT}, the shift due to irradiation ΔRT_{NDT}, and a margin term (M). The shift due to irradiation is the product of a chemistry factor (CF) and a fluence factor using the equation $\Delta RT_{NDT} = (CF)f^{(0.28-0.10 \log f)}$.

Where f is the best estimate neutron fluence in units of 10^{19} neutrons per square centimeter (n/cm^2) at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service in question. The EOL fluence for the vessel beltline material is used in calculating ΔRT_{PTS}.

The margin term is calculated as follows:

$M = \sqrt{\sigma_u^2 + \sigma_\Delta^2}$

Where σ_U is the standard deviation for $RT_{NDT(U)}$. If a measured value of $RT_{NDT(U)}$ is used, then σ_U is determined from the precision of the test method. If a measured value of $RT_{NDT(U)}$ is not available and a generic mean value for that class of materials is used, then σ_U is the standard deviation obtained from the set of data used to establish the mean. If a generic mean value given in paragraph (c)(1)(i)(B) of this section for welds is used, then σ_U is 17 degree Fahrenheit (°F). The value σ_Δ is the standard deviation for ΔRT_{NDT}. The value of σ_Δ to be used is 28 °F for welds and 17 °F for base metal; the value of σ_Δ need not exceed one-half of ΔRT_{NDT}.

The EOL RT_{PTS} values are compared to the PTS screening criteria, which are 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials. For the purpose of comparison with this criterion, the value of RT_{PTS} for the reactor vessel must be evaluated according to the procedures of paragraph (c) of 10 CFR 50.61, for each weld and plate, or forging, in the reactor vessel beltline. RT_{PTS} must be determined for each vessel beltline material using the EOL fluence for that material.

The CF is a function of the copper and nickel content of the material and is determined from tables in 10 CFR 50.61, or a material-specific CF may be determined if credible surveillance data exists for the particular material heat.

The NRC has established requirements and criteria in 10 CFR 50.60 for protecting the reactor pressure vessels (RPVs) of light-water reactors (LWRs) against fracture. The rule requires LWRs to meet the reactor pressure vessel (RPV) materials surveillance program requirements set forth in Appendix H to 10 CFR Part 50.

Appendix H to 10 CFR Part 50 provides the NRC staff's criteria for the design and implementation of RPV material surveillance programs for operating LWRs. The rule, in part, requires RPV surveillance program designs and withdrawal schedules to meet the requirements of the edition of American Society for Testing and Materials (ASTM) Standard Practice E185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactors," that is current on the issue date of the ASME Code to which the RPV was purchased, although later editions of ASTM E185 may be used inclusive of the 1982 Edition of ASTM E185 (ASTM E185-82). The rule also requires proposed RPV surveillance programs to be submitted to the NRC and approved prior to implementation. The applicable criteria in ASTM E185-82 are discussed in Section 3.2.2 of this safety evaluation.

License Condition 2H, "Capsule Withdrawal Schedule," of the BVPS-1 Renewed Facility Operating License states that:

> For the renewed operating license term, all capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation.

The NRC Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," provides guidance for methods acceptable to the NRC staff for determining pressure vessel beltline fluence (ADAMS Accession No. ML010890301).

On September 30, 1997, the NRC issued Administrative Letter (AL) 97-004, "NRC Staff Approval for Changes to 10 CFR Part 50, Appendix H, Reactor Vessel Surveillance Specimen Withdrawal Schedules," to all holders of operating licenses for domestic nuclear power plants (with the exception of those who have ceased operations of their facilities or have certified that fuel has been permanently removed from the reactor). In this AL, the NRC staff summarized the Commission's decision promulgated in Commission Memorandum and Order CLI-96-13, which was issued "In the Matter of the Cleveland Electric Illuminating Company (Perry Nuclear Power Plant, Unit 1)" on December 6, 1996. In this Memorandum and Order, the Commission found that while 10 CFR Part 50, Appendix H, III.B.3, requires prior NRC approval for all withdrawal schedule changes, only certain changes require the staff to review and approve the changes through the NRC's license amendment process (10 CFR 50.90 process). Specifically, only those changes that are not in conformance with ASTM E185 referenced in 10 CFR Part 50, Appendix H, are required to be approved through the license amendment process, whereas, changes that are determined to conform to the ASTM standard only require that the staff document its review and verification of such conformance.

On August 27, 2007, the licensee submitted a license renewal application (LRA) for BVPS-1 and BVPS-2 based on NUREG-1801, Revision 1, "Generic Aging Lessons Learned (GALL)

Report." By letters dated November 5, 2009,[1] the NRC granted license renewal to BVPS-1 for a term expiring January 29, 2036, and to BVPS-2 for a term expiring May 27, 2047. Section XI.M31 of the GALL report includes recommended changes to the surveillance capsule withdrawal schedule to address the period of extended operation.

3.0 TECHNICAL EVALUATION

3.1 RT_{PTS} Values

3.1.1 Licensee Evaluation

In WCAP-18102-NP, Revision 1 (Reference 3), the licensee stated that fluence estimates were performed using the transport methods described in WCAP-14040-A, Revision 4 (ADAMS Accession No. ML050120209), which has been generically approved by the NRC staff based on its adherence to the guidance contained in RG 1.190. The licensee provided, in WCAP-18102-NP, Revision 1, plant-specific dosimetry data, demonstrating agreement between measured and calculated fluence values within ±20%, as recommended in RG 1.190.

Enclosure A to the licensee's October 6, 2017, letter, contains the request for approval of modified RT_{PTS} values. The licensee reevaluated the original $RT_{NDT(U)}$ values for four RPV beltline plates. The licensee used the original Charpy V-notch (C_v) test data points (energy in foot-pounds (ft-lbs) as a function of temperature) and determined a new best-fit curve. The licensee used hyperbolic tangent curve fitting software to determine the new curve. The original curves were hand drawn. The licensee referenced its September 20, 2016, letter (ADAMS Accession No. ML16265A047), which previously reported the revised $RT_{NDT(u)}$ values to NRC. Plate 6903-1 is the limiting material with respect to RT_{PTS}. The licensee indicated that this plate was previously predicted to exceed the PTS screening criteria of 270 °F at 39.6 EFPY.

The licensee stated that the previous RT_{PTS} values for 50 EFPY met the 10 CFR 50.61 PTS screening criteria for beltline and extended beltline materials, with the exception of the limiting material (lower shell plate B6903-1, heat C6317-1). Previously, it was expected that the PTS screening limit of 270 °F for lower shell plate B6903-1 would be reached at 39.6 EFPY. BVPS-1 would not have been able to operate to the end of the license extension period without implementing the requirements of 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events," or a flux reduction measure to manage PTS.

The licensee stated that Appendix E of WCAP-18102-NP documents the updated RT_{PTS} values for 50 EFPY and the changes that were made from the previous analysis of record for BVPS-1 PTS, which was contained in WCAP-15571, Supplement 1, Revision 2, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program" (ADAMS Accession No. ML13151A059).

Appendix E to WCAP-18102-NP states that the following changes and updates to the analysis of record for PTS at BVPS-1, WCAP-15571, Supplement 1, Revision 2, have been incorporated into the calculations contained in Table E-1 of the letter report.

1. Incorporation of the Capsule X results as documented in WCAP-17896-NP, Revision 0 (ADAMS Accession No. ML14288A393), updated reactor vessel fluence values, surveillance capsule irradiated material testing results for Lower Shell Plate B6903-1

[1] ADAMS Accession Nos. ML092670259 and ML092920050.

and Intermediate Shell Longitudinal Welds 19-714 A&B (Heat # 305424) (See Sections 4 and 5), and revised credibility conclusions. [refers to Appendix D of the report]

2. Incorporation of sister plant surveillance capsule test results for weld Heat # 90136 from the Millstone Unit 2 reactor vessel surveillance capsule program (See Sections 4, 5 and Appendix D). Due to the uncertainty in the incorporation of the surveillance data for Millstone Unit 2 (two wire heats were used in the Millstone 2 surveillance weld, with some specimens being Heat # 90136 and others from another weld wire (Heat# 10137), a full-margin term was used for this material in the RT_{PTS} calculations contained in Table 1, even though the revised credibility analysis confirmed that Heat# 90136 remained credible.

3. Incorporation of revised initial reference nil-ductility transition temperature ($RT_{NDT(U)}$) values for the Beaver Valley Unit 1 reactor vessel plate materials as documented in Westinghouse Letter MCOE-LTR-15-15-NP, Revision 1. It was also concluded that the upper shell material for the Beaver Valley Unit 1 reactor vessel has an appropriate $RT_{NDT(U)}$ value even though Branch Technical Position (BTP) 5-3, Paragraph B1.1 (3) methodology for SA-508, Class 2 forging material must be used, due to lack of clear definition of Charpy V-notch orientation. The initial RT_{NDT} value of this material remains drop-weight limited due to the excellent Charpy V-notch test results, as documented in its Certified Material Test Report (CMTR).

4. Utilization of BWRVIP-173-A (ADAMS Package Accession No. ML120830442) to redefine the initial RT_{NDT} values of the six Beaver Valley Unit 1 nozzle materials.

5. Utilization of the following two conclusions from Section 4 of TLR-RES/DE/CIB-013-01 (ADAMS Accession No. ML14318A177), as appropriate:

    a. *The beltline is defined as the region of the RPV adjacent to the reactor core that is projected to receive a neutron fluence level of $1x10^{17}$ n/cm^2 (E > 1.0 MeV) or higher at the end of the licensed operating period.*

    b. *Embrittlement effects may be neglected for any region of the RPV if either of the following conditions are met: (1) neutron fluence is less than $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL, or (2) the mean value of ΔT_{30} estimated using an ETC [Embrittlement Trend Curve] acceptable to the staff is less than 25°F at EOL. The estimate of ΔT_{30} at EOL shall be made using best-estimate chemistry values.*

6. Therefore embrittlement of reactor vessel materials with ΔT_{30} (which is equivalent to ΔRT_{NDT}) values less than 25°F need not be considered in the subsequent RT_{PTS} calculations documented in Table E-1.

3.1.2 Staff Evaluation

Since the licensee used NRC-approved methods to determine the fluence, and since the licensee provided plant-specific data qualifying the methods for use at BVPS-1, the NRC staff determined that the fluence estimates are consistent with RG 1.190, and hence acceptable.

The NRC staff notes that RG 1.190 provides guidance appropriate for fluence evaluations in the RV beltline region. The licensee performed evaluations for extended beltline materials – those that may be a distance from the active fuel region of the core, but whose projected fluence exceeds $1x10^{17}$ n/cm^2 – by using the peak fluence estimated for the same material at a location closer to the core. This is a conservative approach, because fluence decreases as the transport distance from the core increases. Because of this conservatism, the NRC staff accepts the licensee's approach.

The previous analysis of record for PTS for BVPS-1, which the licensee stated as being contained in WCAP-15571, Supplement 1, Revision 2, was not submitted for review and approval by the NRC. The most recent PTS analysis for BVPS-1 reviewed and approved by the staff was in the BVPS LRA, dated August 2007, which the staff documented in Section 4.2.2 of NUREG-1929, Volumn 2, "Safety Evaluation Report Related to the License Renewal of Beaver Valley Power Station, Units 1 and 2" (ADAMS Accession No. ML093000278).

The NRC staff notes that the reduction in the RT_{PTS} value for the limiting plate material B6903-1 is primarily due to the reduction in the $RT_{NDT(u)}$ value from 27° F to 13.1° F.

The licensee cited NRC technical letter report (TLR) TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels," Office of Nuclear Regulatory Research [RES], dated November 14, 2014 (ADAMS Accession No. ML14318A177), as a basis for not considering the shift due to irradiation for RPV materials for which the predicted ΔRT_{NDT} is less than 25 °F. The staff notes that TLR-RES/DE/CIB-2013-01 is not NRC guidance, and the recommendation that the shift due to irradiation can be discounted if it is less than 25 °F is not endorsed in any NRC guidance document or regulation. Therefore, in RAI 1, the staff requested that the licensee revise its submittal to remove reference to the TLR and to apply the appropriate shift to all materials. By letter dated April 6, 2018, the licensee responded to RAI-1 and stated that the PTS evaluation contained in Appendix E of WCAP-18102-NP, Revision 0, has been revised. The licensee provided Revision 1 of WCAP-18102-NP, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation" (ADAMS Accession No. ML18099A125).

The licensee stated that the PTS_{TS} (RT_{PTS}) values for the beltline and extended beltline materials are included in Appendix E of WCAP-18102-NP, Revision 1, and account for the effects of neutron radiation, consistent with 10 CFR 50.61.

Finally, the licensee stated that the BVPS-1 PTS evaluation contained in Appendix E of WCAP-18102-NP, Revision 1, does not utilize the conclusions contained in Section 4 of TLR-RES/DE/CIB-2013-01, and has been revised to remove references to TLR-RES/DE/CIB-2013-01.

The staff checked the RT_{PTS} calculations in Appendix E of WCAP-18102-NP, Revision 1, and verified that these calculations account for the effects of neutron irradiation for all materials, including the materials with a calculated shift of less than 25 °F. The staff, therefore, finds the licensee's response to RAI-1 acceptable.

The NRC staff notes that the licensee's September 20, 2016, letter, indicates that the revised $RT_{NDT(u)}$ values were graphed using computer code CVGRAPH, Version 6.0, with a symmetric hyperbolic tangent curve fit through the minimum data points in accordance with ASME Code, Section III, Subarticle NB-2331, paragraph (a)(4).

Subarticle NB-2331, paragraph (a)(4), states that when a C_v test has not been performed at T_{NDT} + 60, or when the C_V test at T_{NDT} + 60 does not exhibit 50 ft-lbs and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lbs and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed.

The NRC staff reviewed the new C_v curves provided in the licensee's submittal. The new curves were determined for four different RPV plates. Lower shell plate B6903-1 is the plate with the highest 50 EFPY RT_{PTS} value. The $RT_{NDT(u)}$ value of plate B6903-1 decreased from 27 degrees F to 13 °F as a result of the revision of the C_v curves, corresponding to a decrease in the C_v 50 ft-lb temperature from 87 °F to 73 °F. The staff notes that the $RT_{NDT(u)}$ values changed substantially for most of the material heats. The new C_v curves in the licensee's submittal only show one data point at each test temperature. The ASME Code, Section III, NB-2321, specifies that a C_v test consists of a set of three C_v specimens tested at the same temperature. The staff checked the C_V data points in Figures 2 through 9 of the licensee's submittal against the C_V data for the BVPS-1 RPV materials provided in the BVPS-1 response to GL 92-01, dated July 8, 1992 (ADAMS Legacy Accession No. 92077150345), and found that full C_V curves were generated for each beltline material heat, with three specimens being tested at each temperature, and that the licensee based its new curve fits on the lowest C_V energy value at each temperature. Therefore, the NRC staff finds the licensee determined the new $RT_{NDT(u)}$ values in accordance with the ASME Code, Subarticle NB-2331, paragraph (a)(4). The staff plotted the C_V curves using the licensee's curve fitting parameters for each heat, and independently checked the temperature at which the C_V energy is equal to 50 ft-lbs. The NRC staff results agree with the licensee's results. The licensee also provided a table for each plate comparing the input (measured) minimum C_V energy with the computed C_v energy (using the newly determined fit equation) and the differential between them. These differentials were generally less than 5 ft-lbs within the transition region, so the new equations provide a good fit to the data.

Other than for the four plate materials discussed above, the initial $RT_{NDT(u)}$ values, σ_i and σ_Δ values are unchanged from those reported in the LRA. The $RT_{NDT(u)}$ values used are generic in accordance with 10 CFR 50.61, which states that: (i) If a measured value of $RT_{NDT(U)}$ is not available, a generic mean value for the class of material may be used if there are sufficient test results to establish a mean and a standard deviation for the class, and (ii) For generic values of weld metal, the following generic mean values must be used unless justification for different values is provided: 0 °F for welds made with Linde 80 flux, and –56 °F for welds made with Linde 0091, 1092 and 124 and ARCOS B–5 weld fluxes.

For the weld materials, the licensee used a σ_i value of 17 °F since the $RT_{NDT(u)}$ values are generic. The NRC staff, therefore, finds that the licensee's new curve fits for the four plates and provide a reasonable basis for the revised $RT_{NDT(u)}$ values for these plates.

In Appendix E of its submittal, the licensee stated that it determined the $RT_{NDT(u)}$ values for the RPV inlet and outlet nozzle materials using the methodology in BWRVIP-173-A, "BWR Vessel and Internals Project, Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials," and referred to Section 3 and Appendix B of WCAP-18102 for further detail. In Appendix B to WCAP-18102, the licensee elaborated that it employed Alternative Approach 2 found in Appendix B to BWRVIP-173-A. The licensee stated it plotted the C_V curves from the data in the certified material test reports (CMTRs) for the nozzles using CVGraph Version 6.02 (a hyperbolic tangent curve fitting software program). The licensee stated that since the Charpy specimen orientation was not recorded on the CMTRs, it conservatively assumed that the specimens were oriented in the strong direction; thus, the 50 ft-lb transition temperatures were

increased by 30 °F to provide conservative estimates for specimens oriented in the weak direction in accordance with Alternative Approach 2 from BWRVIP-173-A.
The NRC staff reviewed the licensee's determination of the $RT_{NDT(u)}$ values for the inlet and outlet nozzles and determined that it is consistent with Alternative Approach 2 in BWRVIP-173-A. Although the NRC staff approved this methodology specifically for boiling-water reactors (BWRs), there is no reason that it should not be applicable to forgings used in pressurized-water reactors (PWRs) since both use the same material specification (SA-508, Class 2). Therefore, the staff finds the $RT_{NDT(u)}$ values for the inlet and outlet nozzle forgings acceptable. The method of estimating the 50 ft-lb temperature from the strong direction C_v results is similar to, but more conservative than BTP 5-3, Position 1.1(3)(b), which states that "temperatures at which 68 J (50 ft-lbs) and 0.89 mm (35 mils) lateral expansion were obtained on longitudinally-oriented specimens may be increased by 11 °C (20 °F) to provide a conservative estimate of the temperature that would have been necessary to obtain the same values on transversely-oriented specimens.

The licensee stated that copper (Cu) and nickel (Ni) weight percent values, as well as CF values for the nozzle forgings were obtained from WCAP-15571, Supplement 1, Revision 2. Note (g) to Table 4-2, "BVPS-1 Extend Beltline Material Properties," in WCAP-15571, Supplement 1, Revision 2, states that the Cu wt % (weight percent) was not available from the CMTR, so in accordance with RG 1.99, Revision 2, a standard deviation analysis (average + standard deviation) was done to determine the value based on Westinghouse 508 Class 2 Nozzle Forgings (178 data points). Therefore, the NRC staff finds the CF values for the inlet and outlet nozzle forgings acceptable.

The NRC staff performed independent confirmatory calculations of the licensee's RT_{PTS} values provided in Table E-1 of the licensee's submittal. The staff's calculations included an independent determination of the CF for the materials from the tables in RG 1.99, Revision 2, based on the Cu and Ni content of the materials reported by the licensee in Table 3-2 of its submittal. For those materials having surveillance data, the staff performed confirmatory calculations of the best-fit CF from the information provided in Tables 4-1 and 4-2 of the licensee's submittal using the method described in Position 2.1 of RG 1.99, Revision 2. This includes an adjustment for differences in chemistry between the vessel weld and the surveillance weld and an adjustment for differences in irradiation temperature when sister plant surveillance data is used. These tables provided the capsule fluences and the measured 30 ft-lb temperature shifts (measured ΔRT_{NDT}) for these materials. Two of the materials, plate B6903-1 and weld heat 305424, are the BVPS-1 surveillance materials. Surveillance specimens for these two heats were irradiated only in the BVPS-1 reactor. Sister plant surveillance data applicable to BVPS-1 includes weld heat 90136, contained in surveillance capsules at St. Lucie Plant, Unit 1, and Millstone Power Station, Unit 2, and weld heat 305414 which was contained in surveillance capsules at Fort Calhoun Station. The licensee's submittal did not provide the Cu and Ni values for weld 305424. The staff obtained these values from WCAP-17896-NP, Revision 0. "Analysis of Capsule X from the FirstEnergy Nuclear Operating Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program". For weld heat #305414, the NRC staff obtained the Cu and Ni values from "Analysis of Capsule W-275 Omaha Public Power District Fort Calhoun Station Unit No. 1 Reactor Vessel Material Surveillance Program BAW-2226" (ADAMS Accession No. ML12242A050).

For the BVPS-1 RPV beltline materials, the NRC staff's confirmatory calculations yielded identical or very similar RT_{PTS} values to those calculated by the licensee. The staff's calculated RT_{PTS} values for the most limiting material, lower shell plate B6903-1, agree with the licensee's.

This plate has non-credible surveillance data, but the RT_{PTS} calculated using the non-credible surveillance data is less limiting than the RT_{PTS} calculated using the RG 1.99, Revision 2 table CF. Using the RG 1.99, Revision 2 table, the staff calculated an 50 EFPY RT_{PTS} of 258 °F for plate B6903-1, which is less than the applicable PTS screening criteria of 270 °F. Based on the NRC's confirmatory calculations of RTPTS values, the NRC staff concludes that the licensee's revised RT_{PTS} values are acceptable.

3.2 Modification to RV Surveillance Capsule Withdrawal Schedule

3.2.1 Licensee Evaluation

Pursuant to the surveillance program criteria of 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Section II1.B.3, the licensee requested approval of proposed changes to the BVPS-1 RV material irradiation surveillance capsule withdrawal schedule. The licensee stated its request is also submitted to satisfy BVPS-1 renewed operating license condition 2.H, "Capsule Withdrawal Schedule," that states:

> For the renewed operating license term, all capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation.

The licensee stated that the proposed BVPS-1 RV surveillance capsule withdrawal schedule is consistent with the recommendations included in Appendix G, "Surveillance Capsule Withdrawal Schedule," of WCAP-18102-NP, Revision 0. The licensee also stated that the proposed changes to the reactor vessel surveillance capsule withdrawal schedules are consistent with the recommendations specified in ASTM Standard E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," and the end-of-life (EOL) capsule withdrawal requirement in NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," Section XI.M31, "Reactor Vessel Surveillance."

The licensee indicated that a coordinated PWR vessel surveillance program has been developed and is documented in the Electric Power Research Institute technical report, "Materials Reliability Program: Coordinated PWR Reactor Vessel Surveillance Program (CRVSP) Guidelines (MRP-326)" dated 2011 (ADAMS Accession Nos. ML12040A314 and ML12040A315). The purpose of the CRVSP is to increase the fluence levels of future surveillance capsules at withdrawal while maintaining compliance with 10 CFR Part 50, Appendix H, and consistency with the license renewal guidance in NUREG-1801. The licensee also stated that the CRVSP will help generate high-fluence PWR surveillance data in support of extended life operations.

The licensee provided tables showing the current and proposed surveillance capsule withdrawal schedules, as shown below.

Table 1 - Current BVPS-1 Surveillance Capsule Withdrawal Schedule

Capsule	Current (Original) Capsule Location	Lead Factor[(a)]	Withdrawal EFPY[(b)]	Fluence, f[(a)] [n/cm^2, E>1.0 MeV]
V	165°	1.61	1.16	2.99×10^{18}
U	65°	1.06	3.59	6.04×10^{18}
W	245°	1.11	5.89	9.30×10^{18}
Y	295°	1.2	14.29	2.05×10^{19}
X[(c)]	285°	1.72	26.5	5.01×10^{19}[(c)]
T[(d)]	65°	0.99	Standby[(d)]	--
S[(e)]	295°	0.64	Standby[(e)]	--
Z[(f)]	165°	1.24	36.6[(f)]	--

Notes:

a) Actual lead factor and fluence from WCAP-15571, Supplement 1, Revision 2.

b) EFPY from plant startup. Changes to this column will require prior NRC approval (except to indicate that capsules have been removed as specified in Section III.B.3, Appendix H of 10 CFR 50).

c) Capsule X is planned to be withdrawn at the end of Cycle 22, which corresponds to 26.5 EFPY. This capsule will meet the requirements of ASTM E 185-82 for the fifth capsule to be withdrawn for the 40-year EOL.

d) Capsule T was moved to the Capsule U location at the end of Cycle 10. In order to achieve higher fluence data for this capsule, Capsule T should be relocated to the current Capsule Z location when Capsule Z is withdrawn from the vessel [see footnote (f)].

e) Capsule S was moved to the Capsule Y location at the end of Cycle 19. In order to achieve higher fluence data for this capsule, Capsule S should be relocated to the Capsule X location when Capsule X is withdrawn from the vessel at 26.5 EFPY.

f) Capsule Z was moved to the original Capsule V location at the end of Cycle 10. Based on the current information, Capsule Z should be withdrawn after 36.6 EFPY, which corresponds to the peak vessel fluence at 60-year EOL (50 EFPY), 5.58×10^{19} n/cm^2 (E > 1.0 MeV).

Table 2 - Proposed BVPS-1 Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location	Status [(a)]	Capsule Lead Factor[(a)]	Withdrawal EFPY[(b)]	Capsule Fluence[(c)] (n/cm^2, E>1.0 MeV)
V	165°	Withdrawn (EOC 1)	1.47	1.2	2.97×10^{18}
U	65°	Withdrawn (EOC 4)	1.00	3.6	6.18×10^{18}
W	245°	Withdrawn (EOC 6)	1.05	5.9	9.52×10^{18}
Y	295°	Withdrawn (EOC 13)	1.14	14.3	2.10×10^{19}
X	285°	Withdrawn (EOC 22)	1.57	26.6	4.99×10^{19}
S[(d)]	285° (45°/295°)	In Reactor	0.74 [(d)]	Note (d)	2.58×10^{19} [(d)]
T[(e)]	65° (55°)	In Reactor	0.94 [(e)]	Note (e)	3.28×10^{19} [(e)]
Z[(f)]	165° (305°)	In Reactor	1.20 [(f)]	Note (f)	4.18×10^{19} [(f)]

Notes:

(a) Updated in WCAP-18102-NP, Revision 0; Table 2-12.

(b) EFPY from plant startup.

(c) Updated in WCAP-18102-NP, Revision 0; Table 2-11.

(d) Capsule S was moved to the Capsule Y location at the end of cycle 19, and then moved to the Capsule X location at the end of Cycle 22. Reported fluence value and lead factor are accumulated through the end of Cycle 24. Capsule S should remain in the reactor. If additional metallurgical data is needed for BVPS-1, such as in support of a second license renewal to 80 total years of operation, withdrawal and testing of Capsule S should be considered.

(e) Capsule T was moved to the Capsule U location at the end of Cycle 10. Reported fluence value and lead factor are accumulated through the end of Cycle 24. Capsule T should remain in the reactor and continue to accrue irradiation for potential future testing, if needed.

(f) Capsule Z was moved to the original Capsule V location at the end of Cycle 10. Reported fluence value and lead factor are accumulated through the end of Cycle 24. Based on the current information, Capsule Z should be withdrawn after 39 EFPY, which corresponds to the peak vessel fluence at end-of-license extension (50 EFPY), 5.89×10^{19} n/cm^2 (E > 1.0 MeV).

3.2.2 Staff Evaluation

The current RPV surveillance capsule withdrawal schedule was approved by an NRC staff SE dated July 17, 2014 (ADAMS Accession No. ML13242A266). BVPS-1 has withdrawn five surveillance capsules to date. The withdrawal schedule, therefore, is not being changed for these capsules. However, the neutron fluence values at withdrawal have been recalculated, resulting in minor changes to the previous calculated fluence values. Three capsules remain in the RPV, Capsules S, T and Z.

Table 1 of ASTM E185-82 provides the recommended capsule withdrawal schedule as a function of the predicted maximum shift in RT_{NDT} of the RPV materials over the life of the plant. For RPVs with a predicted shift in RT_{NDT} of greater than 200 °F (such as BVPS-1), Table 1 of ASME E185-82 recommends at least five capsules be withdrawn and tested. For the final capsule, Table 1 recommended the capsule fluence be no less than one and no greater than 2 times the maximum predicted RPV inner surface fluence at EOL. Note (f) to the proposed surveillance capsule withdrawal schedule in Table 2 of this SE indicates that Capsule Z should be withdrawn at 39 EFPY, which corresponds to the peak RPV fluence of $5.89 \text{x} 10^{19}$ n/cm^2 at the end–of-license extension of 50 EFPY. The expected fluence at withdrawal for Capsule Z meets the recommendation of Table 1 of ASTM E185-82 for the final capsule, since the fluence will be equal to or greater than one times the expected EOL RPV fluence.

In its safety evaluation report related to the LRA of BVPS-1 and 2 (NUREG-1929), the NRC staff confirmed that the program description of the RPV Integrity Aging Management Program was consistent with AMP XI.M31, Reactor Vessel Surveillance," from NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 1, dated September 2005. The NRC staff also found in NUREG-1929 that the applicant's description of the Reactor Vessel Integrity Program for BVPS-1 and 2, acceptable because the surveillance program design, the capsule withdrawal schedule, and the evaluation of test results are consistent with ASTM E185-82. As capsules are withdrawn from the RV, specimens are stored for future reconstitution, if needed. NUREG-1929 further states that the program manages the remaining standby capsules to achieve the withdrawal of at least one capsule when the capsule neutron fluence is greater than one, but less than two, times the 60-year maximum RV neutron fluence, and that the remaining standby capsules will be managed for optimal neutron exposure and meaningful metallurgical data, if additional license renewals are sought. NUREG-1929 states that the program manages the review and updating of 60-year neutron fluence projections to support the preparation of new P-T limit curves and RT_{PTS} calculations for altered RV exposure conditions.

The staff finds that the changes to the BVPS-1 surveillance capsule withdrawal schedule, specifically the proposal to withdraw Capsule Z at a neutron fluence equivalent to the 60-year (50 EFPY) peak RPV fluence, to be consistent with the staff findings regarding the Reactor Vessel Integrity Program in NUREG-1929, e.g., the capsule withdrawal schedule will remain consistent with the recommendations of GALL AMP XI.M31.

The NRC staff determined that the proposed surveillance capsule withdrawal schedule meets the requirements of ASTM E185-82 that have been incorporated by reference in 10 CFR 50,

Appendix H. The proposed schedule also meets license condition 2.H of the renewed license for BVPS-1.

4.0 CONCLUSION

The NRC staff has reviewed FENOC's proposed PTS values for BVPS-1 for 50 EFPY and has determined that the proposed values will meet the criteria in 10 CFR 50.61.

The NRC staff has reviewed FENOC's proposed surveillance capsule withdrawal schedule for BVPS-1, and finds the proposed schedule to be acceptable because it meets the requirements in 10 CFR Part 50, Appendix H, is in compliance with ASTM E185-82, are consistent with the recommendations of GALL Section XI.M31, and meet the license condition, as described in NUREG-1929, regarding the capsule withdrawal schedules as part of the Reactor Vessel Integrity Program. The NRC staff, therefore, concludes that the proposed RV surveillance capsule withdrawal schedules are acceptable to replace the current surveillance capsule withdrawal schedules.

The NRC staff did not review or approve the P-T limits in WCAP-18102-NP, Revision 0, or in Revision 1 to the same report which was submitted via the licensee's April 6, 2018, letter.

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