DRAFT

REQUEST FOR ADDITIONAL INFORMATION

OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST TO IMPLEMENT 10 CFR 50.61a

"ALTERNATE FRACTURE TOUGHNESS REQUIREMENTS

FOR PROTECTION AGAINST PRESSURIZED THERMAL SHOCK EVENTS"

BEAVER VALLEY POWER STATION UNIT 1

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-334

By letter dated July 30, 2013,[1] FirstEnergy Nuclear Operating Company (FENOC, the licensee), submitted a license amendment request for Beaver Valley Power Station, Unit No. 1 (BVPS-1). The proposed amendment would authorize the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock (PTS) Events," in lieu of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." To complete its review, the Nuclear Regulatory Commission (NRC) staff requests a response to the Request for Additional Information (RAI) questions below.

RAI 1

Background

Under Section 3.4, Neutron Fluence Values, the license amendment request indicates that neutron fluence values at 50 Effective Full Power Years (EFPY) were interpolated from the 48 and 54 EFPY data available in WCAP-15571-NP, Supplement 1, Revision 2, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program ." Regarding WCAP-15571-NP, Supplement 1, Revision 2, the fluence methodology is not described in much detail.

Request

Please provide details of the fluence methodology to enable the NRC staff to verify that the calculations are in accordance with Regulatory Guide 1.190 (RG 1.190),[2] "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML13212A027.
[2] ADAMS Accession No. ML010890301.

RAI 2

Background

When performing the statistical tests required by10 CFR 50.61a, all heats of steels from the RPV of interest for which 10 CFR 50 Appendix H surveillance data exist shall be evaluated. The statistical tests shall be performed separately for each heat. For each heat for which data are available, the ΔT_{30} values used in the statistical tests shall include data from both of the following sources:

- Data obtained for the heat of material in question as part of a 10 CFR 50 Appendix H surveillance program conducted for the plant in question, and
- Data obtained for the heat of material in question as part of a 10 CFR 50 Appendix H surveillance program conducted for any other plant that is operating, or has operated, under a license issued by the NRC. Data from this source is often referred to as having come from a “Sister Plant.”

Issue

The submittal identifies and analyzes data from St. Lucie Plant, Unit 1 for material heat number 90136 and data from Fort Calhoun Station, Unit 1 for material heat number 305414. Staff has identified Millstone Power Station, Unit 2 as having material heat number 90136 in its Appendix H surveillance program and James A. FitzPatrick Nuclear Power Plant as having material heat number 305414 in its Appendix H surveillance program.

Request

Provide the information, analysis and addition of the surveillance materials from Millstone Power Station, Unit 2 (heat 90136), James A. FitzPatrick Nuclear Power Plant (heat 305414), and any other plant-specific or integrated program if the surveillance data satisfy the criteria described in paragraphs 10 CFR 50.61a(f)(6)(i)(A) and 10 CFR 50.61a(f)(6)(i)(B) as they pertain to the statistical tests required by 10 CFR 50.61a.

RAI 3

Background

Paragraph (C)(1) to 10 CFR 50.61a states, “*Each licensee shall have projected values of RT_{MAX-X} for each reactor vessel beltline material for the EOL fluence of the material. The assessment of RT_{MAX-X} values must use the calculation procedures given in paragraphs (f) and (g) of this section. The assessment must specify the bases for the projected value of RT_{MAX-X} for each reactor vessel beltline material, including the assumptions regarding future plant operation (e.g., core loading patterns, projected capacity factors); the copper (Cu), phosphorus (P), manganese (Mn), and nickel (Ni) contents; the reactor cold leg temperature (T_C); and the neutron flux and fluence values used in the calculation for each beltline material.*”

Issue

In Table 3.3-1, “Details of RT_{MAX-X} Calculation Inputs for BVPS-1,” the method of determining $RT_{NDT(u)}$ for Intermediated Shell Plate B6607-1, Intermediate Shell Plate B6607-2, Lower Shell Plate B6903-1 and Lower Shell Plate B7203-2 is identified as Branch Technical Position - Materials Engineering Branch (MTEB) 5-2, “Fracture Toughness Requirements.” The source of the phosphorus content and $RT_{NDT\,(u)}$ values in Table 3.3-1 of the submittal is identified as the Reactor Vessel Integrity Database (RVID). Sources of information must originate from docketed submittals.

Request

a. Identify or provide the reference documents for the phosphorus content and $RT_{NDT(u)}$ values in Table 3.3-1.
b. Identify the specific methodology in MTEB 5-2 and provide the calculations used to determine $RT_{NDT(u)}$ for each of the plate materials in Table 3.3-1.

RAI 4

Background

A series of equations are contained in 10 CFR 50.61a, including:

Equation 7: CRP = B × (1 + 3.77 × Ni1.191) × f(Cu_e,P) × g(Cu_e,Ni,φt_e)

Where:

B = 102.3 for forgings
B = 102.5 for plates in non-Combustion Engineering manufactured vessels
B = 135.2 for plates in Combustion Engineering vessels
B = 155.0 for welds

The "B" coefficient for plates differs depending on whether the plates are from CE-manufactured vessels or non-CE-manufactured vessels.

Issue

Section 3.3, "Plant Specific Material Properties," includes the following statement:

"*The fabrication of the BVPS-1 reactor vessel was initiated by Babcock and Wilcox (B&W) (Reference 6) and completed by Combustion Engineering (CE). In the alternate PTS rule ΔT_{30} correlation, there are two options for plate material coefficient "B", a term used to calculate ΔT_{30}; one for CE manufactured vessels and one for non-CE manufactured vessels. According to the NRC reactor vessel integrity database (RVID), BVPS-1 is considered to be a CE manufactured vessel. However, since B&W purchased, specified the requirements, and did the testing for the plate materials, the non-CE manufactured value for coefficient "B" will be used to calculate ΔT_{30} for the intermediate and lower shell plates.*"

The technical basis for the ΔT_{30} equation in 10 CFR 50.61a and embrittlement analysis data base are described in ORNL/TM-2006/530, (ADAMS Accession No. ML081000630), "A Physically Based Correlation of Irradiation-Induced Transition Temperature Shifts for RPV Steels," and NUREG/CR-6551, (ADAMS Accession No. ML14010A095), "Improved Embrittlement Correlations for Reactor Pressure Vessel Steels," respectively. Appendix A, "Analysis Data Base," to NUREG/CR-6551 identifies BVPS-1as a CE-manufactured vessel. The data from BVPS-1used in development of the ΔT_{30} equation in 10 CFR 50.61a were considered as CE-manufactured. In development of the ΔT_{30} equation in 10 CFR 50.61a, analysis determined that plates in CE-manufactured vessels and non-CE-manufactured vessels would be separated because they had significantly different shifts, and therefore different coefficients for "B" are provided in the ΔT_{30} equation in 10 CFR 50.61a.

Request

Provide a technical basis for the consideration of BVPS-1 as a non-CE-manufactured vessel in the context of the development of the ΔT_{30} equation in 10 CFR 50.61a, in which the BVPS-1 data were considered as CE-manufactured, or revise the ΔT_{30} calculations using the CE-manufactured coefficient for "B."

RAI 5

Background

Sources of information must originate from docketed submittals. The RVID is a spreadsheet of submittal responses to Generic Letter 92-01 and was last updated in the year 2000.

Issue

Section 3.5, "Surveillance Capsule Data," states that "...*the Fort Calhoun Unit 1 surveillance weld data were obtained from the surveillance material baseline evaluation (Reference 11) and the RVID.*"

Request

Identify or provide the reference documents for the Fort Calhoun Unit 1 surveillance data discussed in Section 3.5.

RAI 6

Background

The reactor vessel beltline is defined in 10 CFR Part 50, Appendix G as the region of the reactor vessel that "*directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage*." Appendix H to 10 CFR Part 50 provides the requirements to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline resulting from exposure to neutron irradiation and the thermal environment. Appendix H to 10 CFR Part 50 states that no material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods that the peak neutron fluence at the end of the design life will not exceed 1×10^{17} neutrons/centimeter-squared (n/cm^2) with energy greater than one million electron volts (E > 1 MeV). Appendix G to 10 CFR Part 50 states, "*To demonstrate compliance with the fracture toughness requirements of section IV of this Appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H of this part.*" Therefore, the beltline definition in 10 CFR Part 50, Appendix G is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1×10^{17} n/cm^2 (E > 1 MeV), and this fluence threshold remains applicable for the design life as well as throughout the licensed operating period.

Issue

Paragraph c(1) to 50.61a states, "*Each licensee shall have projected values of RT_{MAX-X} for each reactor vessel beltline material for the EOL fluence of the material. The assessment of RT_{MAX-X} values must use the calculation procedures given in paragraphs (f) and (g) of this section. The assessment must specify the bases for the projected value of RT_{MAX-X} for each reactor vessel beltline material, including the assumptions regarding future plant operation (e.g., core loading* *patterns, projected capacity factors); the copper (Cu), phosphorus (P), manganese (Mn), and nickel (Ni) contents; the reactor cold leg temperature (T_C); and the neutron flux and fluence values used in the calculation for each beltline material.*"

The submittal lacks neutron flux values in Table 3.3-1, and lacks values for Cu, P, Mn, Ni, neutron flux, fluence, and projected values of RT_{MAX-X} for the materials listed in Table 3.8-1.

Request

Provide neutron flux values for each of the materials listed in Table 3.3-1. Provide values for Cu, P, Mn, Ni, neutron flux, fluence, and projected values of RT_{MAX-X} for the materials listed in Table 3.8-1.