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

February 10, 2014

Mr. Eric A. Larson, Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT 2 - REQUEST FOR ADDITIONAL INFORMATION RE: PRESSURE AND TEMPERATURE LIMITS REPORT, REVISION 6 (TAC NO. MF3224)

Dear Mr. Larson:

By letter dated December 9, 2013,[1] FirstEnergy Nuclear Operating Company (the licensee) submitted the Beaver Valley Power Station, Unit 2 (BVPS-2) Pressure and Temperature Limits Report, Revision 6, to the U.S. Nuclear Regulatory Commission (NRC) staff in accordance with Technical Specification 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)." Since the licensee previously relocated the pressure-temperature (P-T) limits for BVPS-2 to a PTLR, and did not change the methodology for generating the P-T limits, the revision to the PTLR was submitted to the NRC for information in accordance with Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." However, during a cursory review, the NRC staff became concerned that the licensee may not have adequately accounted for the reactor pressure vessel nozzle materials in the development of the P-T curves. Therefore, the staff determined that a full review of the PTLR is necessary. To complete its review, the NRC staff requests responses to the enclosed questions.

The draft questions were sent to Mr. Phil Lashley, of your staff, to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed. On February 2, 2014, Mr. Lashley agreed that you would provide a response within 30 days of the date of this letter.

If you have any questions regarding this matter, please contact me at 301-415-4090.

Sincerely,

Jeffrey A Whited

Jeffrey Whited, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

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[1] Agencywide Documents Access and Management System Accession No. ML13344A983.

REQUEST FOR ADDITIONAL INFORMATION

OFFICE OF NUCLEAR REACTOR REGULATION

PRESSURE AND TEMPERATURE LIMITS REPORT, REVISION 6

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT 2

DOCKET NO. 50-412

By letter dated December 9, 2013,[1] FirstEnergy Nuclear Operating Company (the licensee) submitted the Beaver Valley Power Station, Unit 2 (BVPS-2) Pressure and Temperature Limits Report, Revision 6, to the U.S. Nuclear Regulatory Commission (NRC) staff in accordance with Technical Specification 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)." PTLR Figures 5.2-2 through 5.2-6 indicate that intermediate shell plate B9004-1 is the limiting material with respect to the pressure-temperature (P-T) limits for heatup, cooldown, and leak test. However, calculations by the NRC staff based on information provided in the PTLR indicate that the reactor pressure vessel (RPV) inlet nozzles (material ID B9011-1, B9011-2, and B9011-3) may be controlling for certain portions of the P-T limits for heatup, cooldown, and leak test.

In addition, Section 5.2.11 of the PTLR states, in part:

> The pressure-temperature limit curve shown in Figure 5.2-7 was developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head-to-tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop and Code Case N-640.

Comparison of curves in Figure 5.2.7 to the heatup and cooldown curves based on the RPV beltline appears to demonstrate that ferritic components in the RCS outside the RPV would not be limiting with regard to RCS heatup and cooldown; however, it is not clear that such components were explicitly considered in the development of the P-T limits for normal heatup, cooldown, and leak test.

BACKGROUND

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, "Fracture Toughness Requirements," state, in part, that "[t]his appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor

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Enclosure

coolant pressure boundary [RCPB] of light water nuclear power reactors to provide adequate margins of safety..."

In addition, the regulations in 10 CFR Part 50, Appendix G, Paragraph IV.A state, in part, that "[t]he pressure-retaining components of the [RCPB] that are made of ferritic materials must meet the requirements of the ASME Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code], supplemented by the additional requirements set forth in [paragraph IV.A.2, "Pressure-Temperature Limits and Minimum Temperature Requirements"]..."

Therefore, 10 CFR Part 50, Appendix G requires that P-T limits be developed for the entire RCPB, consisting of ferritic RCPB materials in the RPV beltline (neutron fluence $\geq 1 \times 10^{17}$ n/cm^2, E > 1 MeV), as well as ferritic RCPB materials not in the RPV beltline (neutron fluence < 1×10^{17} n/cm^2, E > 1 MeV). Further, 10 CFR Part 50, Appendix G requires that all RCPB components must meet the ASME Code, Section III requirements.

ISSUE

P-T limit calculations for ferritic RCPB components that are not RPV beltline shell materials, may define curves that are more limiting than those calculated for the RPV beltline shell materials. This may be due to the following factors:

1. Some ferritic RCPB components that are not RPV beltline shell materials, such as nozzles, penetrations, and other discontinuities, are complex geometry components that exhibit significantly higher stresses than those for the RPV beltline region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature (RT_{NDT}) for these components is not as high as that of RPV beltline materials that have simpler geometries.

2. Ferritic RCPB components that are not part of the RPV may have initial RT_{NDT} values, which may define a more restrictive lowest operating temperature than the RPV beltline shell materials.

REQUEST

RAI 1 Footnote "C" to Table 5.2-7 of the PTLR states, "as described in Reference 16, the reactor vessel initial RT_{NDT} values for the inlet nozzles are conservatively assigned values. The actual initial RT_{NDT} values for the reactor vessel inlet nozzles are located in BVPS-2 UFSAR [updated final safety analysis report] Table 5.3-1." Discuss why conservatively assigned values are included in Table 5.3-1 instead of the actual values.

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Jeffrey Whited, Project Manager
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