FENOC

FirstEnergy Nuclear Operating Company

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September 1, 2006
L-06-099

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

Subject: Beaver Valley Power Station, Unit Nos. 1 and 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Proposed Alternative to American Society of Mechanical Engineers Code System Leakage Test Requirements
(Request No. BV3-PT-3)

Pursuant to 10 CFR 50.55a(a)(3)(ii), FirstEnergy Nuclear Operating Company (FENOC) hereby requests NRC approval to use the following alternative for the Beaver Valley Power Station (BVPS) Unit No. 1 third and Unit No. 2 second ten-year interval inservice inspection programs.

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI requires hydrostatic pressure testing of Class 1 pressure retaining piping and valves once per ten-year interval. In lieu of these requirements, FENOC plans to perform a system leakage test in accordance with ASME Code Case N-498-1, and proposes alternative visual examinations of components and piping between the Residual Heat Removal System redundant inlet isolation valves during the system leakage test.

Compliance with the ASME Code Section XI requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The proposed alternative provides an acceptable level of quality and safety. The details of the 10 CFR 50.55a request are enclosed.

FENOC requests approval prior to the BVPS Unit No. 1 maintenance and refueling outage, scheduled for September 2007. If there is a reason to believe the alternative may not be found acceptable, please notify FENOC by December 2006 to support planning and scheduling associated with implementation of the ASME Code requirements.

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No new regulatory commitments are contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Gregory A. Dunn, Manager, FENOC Fleet Licensing, at (330) 315-7243.

Sincerely,

Richard L Mende

for James H. Lash

Enclosure: 10 CFR 50.55a Request - Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(ii)

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

Enclosure to Letter L-06-099

10 CFR 50.55A REQUEST BV3-PT-3

Proposed Alternative
In accordance with 10 CFR 50.55a(a)(3)(ii)

--Hardship or Unusual Difficulty
without a Compensating Increase in Level of Quality or Safety--

1.0 ASME CODE COMPONENTS AFFECTED

Beaver Valley Power Station (BVPS) Unit No. 1 and Unit No. 2 Class 1 components and piping between the Residual Heat Removal redundant inlet isolation valves.

At BVPS Unit No. 1 there is one common line, 3 feet -11 inches in length. At BVPS Unit No. 2 there are two lines, one 3 feet in length, and the other 12 feet - 11 inches in length.

Unit No.	Between Valve(s)	And Valve(s)	Segment Description
1	MOV-1RH-700	MOV-1RH-701	14 inch diameter, 1.406 inch wall, seamless A376
2	2RHS-MOV-701A(B)	2RHS-MOV-702A(B)	12 inch diameter, 1.312 inch wall, seamless SA376

2.0 APPLICABLE CODE EDITION AND ADDENDA

American Society of Mechanical Engineers Code (ASME Code) Section XI, 1989 Edition, no Addenda for BVPS Unit No. 1, and Unit No. 2.

3.0 APPLICABLE CODE REQUIREMENT

Table IWB-2500-1, Category B-P, Item B15.51 requires hydrostatic testing of Class 1 pressure retaining piping once per ten-year interval. Code Case N-498-1 (referenced in the BVPS Ten-Year Inservice Inspection Program) allows a system leakage test in lieu of the ten-year hydrostatic testing. Note 2 of Table IWB-2500-1 and Paragraph (a)(2) of N-498-1 require that the test pressurization boundary extend to all Class 1 components.

Paragraph IWB-5221(a) states, "The system leakage test shall be conducted at a test pressure not less than the nominal operating pressure associated with 100% rated power."

4.0 REASON FOR REQUEST

Background Information

The function of the Residual Heat Removal (RHR) System is to remove decay heat energy from the reactor core to cool the plant for maintenance or refueling. It is designed to perform its function only after the Reactor Coolant System (RCS) temperature and pressure have been reduced to below RHR design limits.

The portions of piping between the inlet isolation valves have the same design pressure and temperature as the RCS (2485 psig, 650°F). However, this piping, along with the remainder of RHR System is isolated from the RCS before the RCS pressure exceeds the design pressure of the remainder of the RHR System. The RHR System is isolated from the RCS by two motor-operated valves in series on each pump suction line. The Class 1 boundary extends to the second isolation valve from the RCS. Each valve is interlocked to prevent its opening if the RCS pressure is greater than 430 psig (BVPS Unit No. 1) or 360 psig (BVPS Unit No. 2) and to automatically close if RCS pressure exceeds 630 psig (BVPS Unit No. 1) or 700 psig (BVPS Unit No. 2).

Prior to the RCS reaching normal operating pressure and temperature (2235 psig and 547°F), the piping segments listed above are isolated from the RCS by closing the inlet motor operated isolation valves in accordance with plant procedures. Therefore, the pressure and temperature of this piping is less than full RCS pressure and temperature.

Hardship or Unusual Difficulty

The plant design configuration complies with the RCS boundary requirements for double valve isolation as defined in 10 CFR 50.2, but cannot satisfy the code test requirement for nominal operating pressure associated with 100 percent rated power. Opening the upstream isolation valve with full RCS pressure and temperature conflicts with the double isolation philosophy of 10 CRF 50.55a(c)(ii) between the Class 1 and Class 2 portions of the RHR System.

The use of temporary hoses is possible at BVPS Unit No. 2 only, to pressurize these sections to full pressure. The BVPS Unit No. 1 segment has no vent or drain connections that would allow the use of temporary hoses. However, the use of temporary hoses is not feasible or safe. Temporary hoses are not qualified to meet all aspects of the plant design (for example, pressure, temperature, ASME code requirements, seismic, and occasional loading). Since the hoses cannot be fully qualified, use of temporary hoses would be unsafe from both a personnel and a nuclear safety perspective. Failure of the hoses could cause injury to test personnel and could cause loss of reactor coolant.

Another alternative is implementation of a plant design change to install qualified piping to allow pressurization between the isolation valves. This alternative is considered cost prohibitive without a compensating increase in the level of quality and safety.

Based on the reasons noted above, FirstEnergy Nuclear Operating Company believes that compliance with the specified code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE

Proposed Alternative

The pipe segments noted above are visually examined (VT-2) for leakage each refueling outage as part of the normal Class 1 system leakage test with the valves in their normal system alignment. This examination is proposed as an alternative to the applicable code requirement identified above.

Additionally, it is proposed that these pipe segments be visually examined (VT-2) as part of the Class 2 system functional test performed once every 40-month period using the Class 2 test requirements.

Basis for Use

If through-wall leakage were to occur in these piping segments, the proposed alternative examinations would identify the leakage.

These pipe segments contain stainless steel pipe, valves and weld material. There are no alloy 600/82/182 materials in these pipe segments. There are no known active degradation mechanisms at work in these pipe segments.

The upstream extensions of these segments, beyond the inlet isolation valves are constructed using the same material specifications. These extensions are exposed to the higher reactor coolant pressure and are included in the Class 1 system leakage test performed every refueling outage.

6.0 DURATION OF PROPOSED ALTERNATIVE

The alternative is requested to be implemented during the final 40-month period of the third ten-year inservice inspection interval at BVPS Unit No. 1 and the second ten-year inservice inspection interval at BVPS Unit No. 2.

7.0 PRECEDENT

The NRC has previously approved a similar request, as demonstrated in the correspondence listed below. The relief request for Beaver Valley Power Station is similar to this precedent, in that FENOC proposes to visually examine the noted piping segments for evidence of leakage each refueling outage as part of the normal Class 1 system leakage test with valves in their normal system alignment. In this configuration the noted piping segments would not be pressurized to the nominal reactor coolant system operating pressure. Additionally, it is proposed that the noted pipe segments be visually examined for evidence of leakage as part of the Class 2 system functional test performed once every 40-month period using the Class 2 test requirements. As stated in the correspondence listed below, the NRC staff concluded that to pressurize the subject piping segments in accordance with the ASME Code requirements would require significant plant modifications and would subject the licensee to an undue burden with no compensating increase in quality or safety.

- Surry Power Station, Unit Nos. 1 and 2, Docket Nos. 50-280 and 50-281, American Society of Mechanical Engineers, Section XI Fourth 10-Year Inservice Inspection Program (TAC Nos. MC5587 and MC5597, Relief Request Nos. SPT-006 for Unit No. 1 and SPT-005 for Unit No. 2), dated November 1, 2005.