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

March 2, 2017

Mr. Marty L. Richey, 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 NO. 1 – RELIEF FROM THE REQUIREMENTS OF THE ASME CODE (CAC NO. MF8531)

Dear Mr. Richey:

By letter dated October 24, 2016, as supplemented by letter dated January 13, 2017 (Agencywide Documents Access and Management System Accession Nos. ML16298A289 and ML17013A483, respectively), FirstEnergy Nuclear Operating Company (FENOC or the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for the use of alternatives to certain American Society of Mechanical Engineers (ASME) Code for Operating and Maintenance of Nuclear Power Plants (OM Code) requirements at Beaver Valley Power Station, Unit No. 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested to use the proposed alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. The proposed alternative is to delay a relief valve test to the maintenance and refueling outage of the spring of 2018.

The NRC staff has reviewed the subject request and determined, as set forth in the enclosed safety evaluation, that FENOC has adequately addressed all of the regulatory requirements set for in 10 CFR 50.55a and that the proposed alternative provides reasonable assurance that the affected component is operationally ready. Accordingly, the NRC staff concludes that FENOC has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2) and is in compliance with ASME OM Code requirements. Therefore, the NRC staff authorizes the proposed alternative request to test valve RV-1RH-721 during the next refueling outage, which is currently scheduled for the spring of 2018.

All other ASME OM Code requirements for which relief was not specifically requested and approved in this relief request remain applicable.

If you have any questions, please contact the Project Manager, Taylor Lamb, at 301-415-7128 or Taylor.Lamb@nrc.gov.

Sincerely,

Stephen S. Koenick, Acting Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE VRR6

REGARDING RESIDUAL HEAT REMOVAL SYSTEM VALVE TESTING

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

1.0 INTRODUCTION

By letter dated October 24, 2016, as supplemented by letter dated January 13, 2017 (Agencywide Documents Access and Management System Accession Nos. ML16298A289 and ML17013A483, respectively), FirstEnergy Nuclear Operating Company (FENOC or the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for the use of alternatives to certain American Society of Mechanical Engineers (ASME) Code for Operating and Maintenance of Nuclear Power Plants (OM Code) requirements at Beaver Valley Power Station, Unit No. 1 (BVPS-1).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested to use the proposed alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. The proposed alternative is to delay a relief valve test to the maintenance and refueling outage of the spring of 2018.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(f), "Inservice testing requirements," require, in part, that inservice testing (IST) of certain ASME Code Class 1, 2, and 3 components must meet the requirements of the ASME OM Code and applicable addenda, except where alternatives have been authorized pursuant to 10 CFR 50.55a(z)(1) or 10 CFR 50.55a(z)(2).

In proposing alternatives, a licensee must demonstrate that the proposed alternatives provide an acceptable level of quality and safety (10 CFR 50.55a(z)(1)), or compliance would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety (10 CFR 50.55a(z)(2)).

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

Enclosure

3.0 TECHNICAL EVALUATION

3.1 Licensee Relief Request No. VRR6

Alternative testing is requested for the BVPS-1 residual heat removal (RHR) pump relief valve RV-1RH-721 (Class 2, Category C).

Applicable ASME OM Code requirements, as stated by the licensee, are:

> Mandatory Appendix I, I-1350, "Test Frequency, Class 2 and 3 Pressure Relief Valves," Paragraph (c), "Requirements for Testing Additional Valves," states in part:
>
> Additional valves shall be tested in accordance with the following requirements:
>
> 1. For each valve tested for which as-found set-pressure (first test actuation) exceeds the greater of either the ± tolerance limit of the Owner established set-pressure acceptance criteria of I-1310(e) or ±3% of valve nameplate set-pressure, two additional valves shall be tested from the same valve group.

The licensee states:

> **Reason for Request**
>
> Charging system letdown relief valve RV-1CH-203 has a cold differential test pressure (CDTP) of 606 pounds per square inch gauge (psig) with an upper acceptable limit of plus 3 percent or 624 psig. The CDTP is a temperature compensated test pressure used to account for the difference between the ambient temperature at the test stand and the higher temperature at the tested valve's installed location in the plant. During the 24th Beaver Valley Power Station, Unit No. 1 (BVPS-1), maintenance and refueling outage (1R24) when RV-1CH-203 was tested on the test stand at ambient conditions, the relief valve lifted at 631.7 psig, which was 4.25 percent above its CDTP. This exceeded the plus 3 percent limit specified in Paragraph I-1350(c)(1) requiring the only other valve in the sample group, RV-1RH-721, to be tested. However, because the ASME OM Code, Appendix I, Paragraph I-1350(c)(1) allows for the owner to establish set-pressure acceptance criteria (a plus or minus tolerance limit), a limit of plus 5 percent was calculated after the valve exceeded the plus 3 percent limit specified in the ASME OM Code. Relief valve RV-1CH-203 test results were within the plus 5 percent acceptance criteria. Therefore, valve RV-1RH-721 was not tested. When it was determined that FENOC was not permitted to provide an owner specified limit after the valve was tested, the plant lineup was being established for plant startup with the reactor core fully loaded with fuel.
>
> Testing of RV-1RH-721 cannot be performed in-place. The relief valve must be removed from the system and tested on a test stand. In order to remove and test RV-1RH-721, the RHR system would have to be shut down and the entire

system drained. This is normally performed when the RHR system is not required to be in operation, which is when fuel is removed from the reactor core.

Based on the plant lineup and the need to remove the valve from the system to perform the test, the sample size could not be expanded as required and is not in compliance with the ASME OM Code. Therefore, a delay is proposed to test RV-1RH-721 during the 25th BVPS-1 maintenance and refueling outage (1R25) when fuel is removed from the core (1R25 is scheduled for the spring of 2018). To unload the reactor core in order to test RV-1RH-721 provides a hardship without a compensating increase in quality and safety.

Proposed Alternative and Basis for Use

The proposed alternative is to delay testing of relief valve RV-1RH-721 until 1R25. At that time, the RHR system can be drained, and RV-1RH-721 can be tested.

Valve RV-1RH-721 is considered operationally ready until tested during 1R25 for the following reasons:

1) A work order history review shows that two different serial-numbered (S/N) valves have been swapped in and out of the installed location for RV-1-RH-721 as follows:

1R11 (April 1996)	S/N 53237M1 installed (set at 610 psig)
1R13 (March 2000)	S/N 53237M1 removed (as-found set at 616 psig), and S/N N69973-01-0001 installed (set at 608 psig)
1R15 (March 2003)	S/N N69973-01-0001 removed (as-found set at 510 psig due to leakage), and S/N 53237M1 installed (set at 610 psig)
1R17 (January 2006)	S/N 53237M1 removed (as-found set at 603 psig), and S/N N69973-01-0001 installed (set at 610 psig)
1R20 (October 2010)	S/N N69973-01-0001 removed (as-found set at 611.2 psig), and S/N 53237M1 installed (set at 606 psig)

The setpoint testing listed above was performed on a test stand at ambient conditions.

The preceding data shows that valve S/N N69973-01-0001 has drifted high approximately 2 psig over a 3 to 4 year period while valve S/N 53237M1 has drifted high approximately 6 psig over a 4 year period. If the approximate 6 psig

high drift for valve S/N 53237M1 is extrapolated over a 7.5 year period from 1R20 to 1R25, RV-1RH-721 would be expected to lift no higher than 618 psig. Therefore, based on past performance, there is reasonable assurance that RV-1RH-721 would not lift greater than plus 3 percent above its set-pressure when tested during the next maintenance and refueling outage.

2) Other than a three-year period from 1R13 to 1R15 when the valve was found to lift low due to leakage, RV-1RH-721 has shown a history of consistent performance. Therefore, there is reasonable assurance that valve RV-1RH-721 will continue to be operationally ready until the next scheduled test during 1R25. Further, the test interval from 1R20 to 1R25 (7.5 years) is conservatively shorter than the maximum 10-year test interval requirement of ASME OM Code, Mandatory Appendix I, Paragraph I-1350(a).

In addition, during operation at power, a surveillance verification log monitors annunciator window A1-125, "Residual Heat Removal Pump Discharge Pressure HIGH." This annunciator alarms if pressure reaches 550 psig and provides corrective actions to take in accordance with an alarm response procedure. The corrective actions would relieve pressure before it reaches the RV-1RH-721 set-pressure of 600 psig.

The proposed alternative to ASME OM Code, Mandatory Appendix I, paragraph I-1350(c) would delay the test of relief valve RV-1RH-721 until 1R25. Therefore, there is reasonable assurance that relief valve RV-1RH-721 will continue to be operationally ready until 1R25.

The proposed alternative is requested for use during the remainder of the fourth 10-year IST interval.

3.2 <u>NRC Staff Evaluation</u>

ASME OM Code, 2001 Edition through 2003 Addenda, Section Mandatory Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," paragraph I-1350(a), 10-year test Interval states that:

> Class 2 and 3 pressure relief valves, with the exception of PWR [pressurized-water reactor] main steam safety valves, shall be tested every 10 years, starting with initial electric power generation. No maximum limit is specified for the number of valves to be tested during any single plant operating cycle; however, a minimum of 20% of the valves from each valve group shall be tested within any 48-month interval. This 20% shall consist of valves that have not been tested during the current 10-year test interval, if they exist.

Requirements for testing additional valves in a group is detailed in paragraph I-1350(c)(1), which states that, "For each valve tested for which the as-found set-pressure (first test actuation) exceeds the greater of either the ± tolerance limit of the Owner established set-pressure acceptance criteria of I-1310(e) or ±3% of valve nameplate set-pressure, two additional valves shall be tested from the same valve group."

As noted in the alternative request, relief valve RV-1CH-203 was recently tested pursuant to the required ASME OM Code test interval and failed to meet the standard ASME OM Code tolerance acceptance value of ±3 percent of valve nameplate set-pressure. The standard ASME OM Code tolerance acceptance value was used because the licensee did not have an owner-established set-pressure acceptance criteria specified. The valve failure now required additional valves of the valve group to be tested. A valve group is defined in ASME OM Code, 2001 Edition through 2003 Addenda, as "Valves of the same manufacturer, type, system application, and service media." Valve RV-1CH-203 has only one other valve associated with it in its group, which is RV-1RH-721.

Relief valve RV-1RH-721 protects the RHR system. The valve cannot be tested in situ and must be removed from the system. To remove the valve from the system will require draining the RHR system, which is normally performed when the fuel is removed from the reactor core. The next scheduled fuel removal is the spring of 2018. To perform the required ASME OM Code test now would represent a hardship, without a compensating increase in quality and safety.

The licensee proposes to delay testing RV-1RH-721 until the next refuel outage when the RHR system can be removed from service after the reactor core is off-loaded. The NRC staff reviewed the performance history of the setpoint testing. Considering the sample, in which there was no instrument drift high above 1 percent, the NRC staff has reasonable assurance that the valve would not lift greater than +3 percent. Furthermore, the NRC staff credits the fact that at power, a surveillance verification log monitors annunciator window A1-125 and that the annunciator alarms if pressure reaches 550 psig. The resulting corrective actions would relieve pressure before they reach the RV-1RH-721 set-pressure of 600 psig. Therefore, the NRC staff concludes that the proposed alternative is acceptable.

4.0 CONCLUSION

As set forth above, the NRC staff determined that the proposed alternative provides reasonable assurance that the affected component is operationally ready. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(z)(2).

All other ASME OM Code requirements for which relief was not specifically requested and approved in the subject requests for relief remain applicable.

Therefore, the NRC staff authorizes the BVPS-1 proposed alternative request to test valve RV-1RH-721 during the next refueling outage, which is currently scheduled for the spring of 2018.

Principal Contributor: M. Farnan

Date: March 2, 2017

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 – RELIEF FROM THE REQUIREMENTS OF THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS CODE (CAC NO. MF8531) DATED MARCH 2, 2017

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