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

March 11, 2013

LICENSEE: FirstEnergy Nuclear Operating Company

FACILITY: Beaver Valley Power Station, Units 1 and 2

SUBJECT: SUMMARY OF FEBRUARY 20, 2013, MEETING WITH FIRSTENERGY NUCLEAR OPERATING COMPANY TO DISCUSS REQUEST FOR ADDITIONAL INFORMATION RESPONSE REGARDING BEAVER VALLEY POWER STATION END-OF-LIFE MODERATOR TEMPERATURE COEFFICIENT TESTING (TAC NOS. ME9144 AND ME9145)

On February 20, 2013, a Category 1 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) and representatives of FirstEnergy Nuclear Operating Company (FENOC, the licensee). The purpose of the meeting was to discuss a request for additional information (RAI) sent by the NRC staff to FENOC by letter dated December 28, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12340A256). The RAI relates to a proposed license amendment request submitted by FENOC (ADAMS Accession No. ML12208A309), to the Beaver Valley Power Station, Units 1 and 2, Technical Specifications (TSs). The proposed amendment would modify TS 3.1.3 to allow the normally required near-end-of-life Moderator Temperature Coefficient measurement to be replaced by a calculated value under certain conditions.

The meeting was open to the public for observation, although portions were closed due to the discussion of information that was proprietary in nature. As indicated in the meeting notice dated February 4, 2013 (ADAMS Accession No. ML13023A175), the response to RAI number 1 was discussed in the open portion of the meeting and the responses to RAI numbers 2 and 3 were discussed in the closed portion. This meeting summary discusses the open portion of the meeting only. A copy of the meeting handout providing a non-proprietary draft response to RAI numbers 1 and 3 is provided in Enclosure 1.

Regarding RAI number 1, the NRC staff asked the licensee to add clarification that the data and conclusions provided in the draft RAI response were based on predictions solely from the PHOENIX-P code. In light of the licensee's proposal that would also permit use of the PARAGON code for predicting the end-of-life moderator temperature coefficient, the NRC staff further requested that the licensee include adequate technical basis to justify extending the conclusions made in the RAI response regarding the PHOENIX-P code to the PARAGON code.

No members of the public participated in the meeting and no meeting feedback forms were received by the NRC staff. A list of attendees is provided in Enclosure 2.

Please direct any inquiries to me at 301-415-2833 or peter.bamford@nrc.gov.

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Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Meeting Handout
2. List of Attendees

cc w/encl: Distribution via ListServ

Enclosure 1

Draft Response to Request for Additional Information Regarding End-of-Life Moderator Temperature Coefficient Testing

By letter to the Nuclear Regulatory Commission (NRC) dated July 25, 2012, FirstEnergy Nuclear Operating Company (FENOC) submitted a proposed amendment to the Beaver Valley Power Station, Units 1 and 2, Technical Specifications (TSs). The proposed amendment would modify TS 3.1.3 to allow the normally required near-end-of-life (EOL) Moderator Temperature Coefficient (MTC) measurement to not be performed under certain conditions. On December 28, 2012 (Accession No. ML1 2340A256) the Nuclear Regulatory Commission (NRC) requested additional information regarding this LAR. The NRC staff's requests are presented in bold type, followed by the draft responses.

1. In accordance with the second condition in the NRC staff's safety evaluation for WCAP-13749-P-A, the licensee proposed to confirm, on a cycle-specific basis, that core fuel design changes or data from MTC predictions and measurements do not show a significant effect on the predictive correction. Please clarify the process and criteria for making this determination and justify their adequacy (e.g., statistical testing, engineering judgment, etc).

Draft Response:

As described in WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," approved in March 1997, the HFP predictive correction accounts for the observed differences between the measured and predictive (M-P) MTCs. "The hot full power (HFP) predictive correction (-3.0 pcm/°F) was "derived by summing the HZP [hot zero power] predictive correction, the xenon sensitivity and the burnup sensitivity." The HZP predictive correction is provided in WCAP-13749-P-A. As long as the beginning of life (BOL) HZP MTC (M-P) is less negative than the HZP predictive correction, the HFP predictive correction is valid for use during the cycle.

The tables below provide BOL HZP MTC measured values (MTC M), predicted values (MTC P), and the measured minus the predicted values (M-P) for each cycle listed for Beaver Valley Power Station (BVPS), Unit 1 and Unit 2. The MTC M for both BVPS-1 and BVPS-2 are consistently more positive than MTC P, and therefore is conservative for evaluating the continued use of the HFP predictive correction value of -3.0 pcm/°F.

Table 1: BVPS-1 BOL HZP MTC Data (all values in pcm/°F)

Cycle	MTC M	MTC P	(M-P)
14	[redacted]		0.035
15			0.166
16			0.035
17			0.067
18			0.064
19			0.361
20			0.423
21			0.254
22			0.473

Table 2: BVPS-2 BOL HZP MTC Data (all values in pcm/°F)

Cycle	MTC (M)	MTC (P)	(M-P)
9	[redacted]		0.38
10			0.088
11			0.244
12			0.238
13			0.192
14			0.748
15			0.638
16			0.759
17			0.693

WCAP-13749-P-A states, "...the [HFP] predictive correction is reexamined if changes in core fuel designs or continued MTC calculation/measurement data show significant effect on the predictive correction." During the BVPS core design process for each cycle, Westinghouse would provide the HFP predictive correction to FENOC in the Nuclear Design Report. FENOC would verify that the predictive correction remains valid for the applicable fuel cycle by performing the following two qualitative assessments.

1. FENOC would assess the effects of fuel and core design methodology changes on the BOL HZP MTC prediction by using the "Fuel Cycle Process" procedure. Prior to each reload, the "Reload Risk Evaluation Checklist" is used to identify and determine the risk of major fuel design changes or core design methodology changes. This checklist identifies whether the reload will use revised or different methodologies, and assesses the impact of these changes on the existing analyses. Additionally, prior to accepting the Westinghouse core design calculations, FENOC compares the results generated by Westinghouse computer models to results generated by FENOC using an independent computer model. These evaluations would provide initial indication of a possible change in the BOL HZP MTC (M-P) relationship prior to startup of the fuel cycle.

2. Per TS 3.1.3, each cycle during low power physics testing, FENOC measures the BOL HZP MTC. Prior to each conditional exemption of the end of life (EOL) HFP MTC measurement test, FENOC would compare BVPS specific MTC (M-P) data each cycle against previous cycles to determine if there is a change to the measured vs. predicted MTC relationship.

   If the value of the BOL HZP MTC (M-P) approaches the HZP predictive correction given in WCAP-13749-P-A then FENOC would evaluate the use of the HFP MTC predictive correction to show that the value of -3.0 pcm/°F is conservative or measure the EOL HFP MTC in accordance with the Technical Specifications.

2. The predictive correction term defined in WCAP-13749-P-A is based, in part, on a tolerance limit that Westinghouse derived from differences between a set of measured and predicted values of the MTC at the beginning of an operating cycle at hot, zero power conditions. Specifically, the predicted MTC values in WCAP-13749-P-A- were determined from calculations using the PHOENIX-P/ANC code package for a variety of pressurized-water reactor (PWR) core designs prior to 1985. Although the NRC staff has approved the PARAGON lattice physics code as a replacement to PHOENIX-P, it cannot not be concluded that the statistical database, and hence the predictive corrections terms, for the two codes will be equivalent. Therefore, if approval for the use of the predictive correction term for the PHOENIX-P code for calculations with the PARAGON code is sought under this license amendment request, please provide unbiased and statistically

significant data analogous to that reported in Table 3-1 of WCAP-13749-P-A for calculations performed with the PARAGON code for contemporary PWR core designs, along with : (1) justification that this data belongs to the same population as the pre-1995 data in WCAP-13749-P-A, generated with PHOENIX-P code; or (2) a new predictive correction term for the PARAGON code for contemporary cores that is based on a 95/95 tolerance limit appropriate for modifying end-of-cycle MTC predictions made with this code.

Draft Response:

To be provided.

3. The staff found the benchmark criteria for application of the conditional exemption methodology in Table 3-2 of WCAP-13749-P-A to be acceptable for the PHOENIX-P code. However, due to the expectation that different codes will lead to different measured-minus-predicted deviation statistics for parameters such as the beginning-of-cycle, hot zero-power isothermal temperature coefficient (BOC HZP ITC) and core reactivity difference, it is not clear that the criteria in Table 3-2 are adequate for the PARAGON code. In essence, the criteria of Table 3-2 should be capable of ensuring that a modified prediction of the MTC is substituted for a measured value only when a given core is observed to behave in a way that is represented by the data used to derive the predictive methodology. Similarity to past data serves as an inductive argument that a given core's characteristics are predictable. Therefore, it is critical that adequate criteria are established to discriminate when the conditional exemption may be permitted based on similarity to existing data for representative core designs. It is further clear that this determination may be code-dependent. Therefore, if approval for the use of the conditional exemption criteria in Table 3-2 of WCAP-13749-P-A with the PARAGON code is sought under this license amendment request, please provide justification that these criteria are appropriate based on unbiased and statistically significant data for BOC HZP ITC and core reactivity difference calculations performed with the PARAGON code for contemporary PWR core designs; alternately, please propose new criteria that are acceptable for predictions using the PARAGON code.

Draft Response:

There are seven benchmark criteria cited in WCAP-13749-P-A, Table 3-2, "Benchmark Criteria for Application of the 300 PPM MTC Conditional Exemption Methodology." Four of these criteria are taken directly from the American National Standards Institute (ANSI) standard 19.6.1, Table A.1: (i) BOC, HZP, ITC; (ii) Individual Control Bank Worth; (iii) Total Control Bank Worth; and, (iv) Assembly Power (Measured Normalized

Reaction Rate). The fifth criterion, Core Reactivity Difference, listed in WCAP-13749-P-A, Table 3-2 is taken directly from NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The sixth and seventh criteria, Measured Incore Quadrant Tilt (Low Power) and Measured Incore Quadrant Tilt (Full Power), are supplementary Westinghouse recommendations for plant operations. WCAP-13749-P-A, Table 3-2 criteria apply not only during startup physics testing, but for routine surveillances when performed throughout the cycle.

WCAP-13749-P-A, Table 3-2 benchmark criteria were originally based on the 1985 ANSI 19.6.1, "Reload Startup Physics Tests for Pressurized Water Reactors," standards. These criteria have the same values as in the 2011 ANSI 19.6.1 standards demonstrating that these criteria have been shown to be robust over an extended period of reactor operations.

The Introduction to the 2011 ANSI 19.6.1 states:

> In conjunction with each refueling shutdown or other significant reactor core alteration, nuclear design calculations are performed to ensure that the reactor physics characteristics of the new core will be consistent with the safety limits. Prior to return to normal operation, successful execution of a physics test program is required to determine if the operating characteristics of the core are consistent with the design predictions and to ensure that the core can be operated as designed.

The ANSI 19.6.1 standards were designed to be independent of the computer code(s) used for design predictions. The ANSI 19.6.1 standards are used to confirm that the predictions produced by the codes are sufficiently accurate and precise in order to provide confidence that design predictions that would not or could not be subjected to testing are also adequate to permit core operation within the safety limits.

The Core Reactivity Difference criterion, taken directly from TSs, is independent of the computer code(s) used to generate the predictions to which the measurements are compared. The purpose of the comparison is to demonstrate that the predictions produced by the computer codes provide confidence that design predictions that either would not or could not be subjected to testing, are adequate to permit core operation within the safety limits.

Since the TSs do not have a specific criterion on incore quadrant tilt for the flux symmetry test required by ANSI 19.6.1 at the beginning of the cycle, Measured Incore Quadrant Tilt (Full Power) and Measured Incore Quadrant Tilt (Low Power) criteria are derived from supplementary guidance supplied by Westinghouse. These criteria were established independent of the computer code(s) used for generating the core design predictions.

A review of the bases for each of the benchmark criteria provided in Table 3-2 has shown that the criteria are either industry standards, a TS, or Westinghouse recommendations for plant operation, that are independent of the computer codes used to generate the design predictions. Section A.3, "Typical criteria and bases," of ANSI 19.6.1 states:

"The criteria are not established by rigorous analysis of the test methods or design models." Instead, "(t)he criteria are established by differences between calculations and measurements that would suggest a problem with the as-built core, the measurement, or the prediction."

Thus, the continuing applicability of the benchmark criteria for application of the 300 PPM MTC conditional exemption methodology is demonstrated by the continuing applicability of the ANSI 19.6.1 standards, TSs, and Westinghouse recommendations for plant operation, and therefore should not be interpreted as being predicated upon any characteristics of the computer code(s) used to generate any specific design predictions.

LIST OF ATTENDEES

NRC/FENOC MEETING TO DISCUSS PROPOSED REQUEST FOR ADDITIONAL INFORMATION RESPONSE REGARDING END-OF-LIFE MODERATOR TEMPERATURE COEFFICIENT TESTING

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

FENOC

Tom Lentz
Julie Hartig
Phil Lashley
A.R. Burger
Bob Huston

NRC

Meena Khanna
Peter Bamford
John Lehning

Westinghouse, FENOC Contractor

Mohammed Ouisloumen
Frank Boylan
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/ra/

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Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Meeting Handout
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RidsOgcRp Resource
JCassidy, EDO Region I
RidsNrrDssSnpb Resource
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ADAMS Accession Nos.: Package ML13057A599; Meeting Notice ML13023A175; Meeting Summary ML13057A678
*via email

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