**FENOC**™
FirstEnergy Nuclear Operating Company

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L-12-223

10 CFR 50.59(d)(2)

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

SUBJECT:
Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Report of Facility Changes, Tests and Experiments

In accordance with 10 CFR 50.59(d)(2), a report of facility changes, tests, and experiments for the Beaver Valley Power Station, Unit No. 1 is attached. The report covers the period July 17, 2010 to May 31, 2012.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Phil H. Lashley, Supervisor - Fleet Licensing, at (330) 315-6808.

Sincerely,

Paul A. Harden

Attachment:
Beaver Valley Power Station, Unit No. 1
Report of Facility Changes, Tests and Experiments

cc: NRC Region I Administrator
NRC Senior Resident Inspector
NRR Project Manager
Director BRP/DEP
Site BRP/DEP Representative

Change Title

Spent Fuel Pool Criticality Analysis, Revised Code Bias Uncertainty

Description

Nuclear Regulatory Commission (NRC) Information Notice 2011-03, "Non-conservative Criticality Safety Analyses for Fuel Storage," identified a non-conservative element in the methodology to determine the Monte Carlo code benchmarking bias uncertainty used in spent fuel pool criticality analyses. The Beaver Valley Power Station, Unit No. 1 spent fuel pool criticality analysis utilized this non-conservative methodology element for spent fuel pool Region 3. Specifically, the statistical review of the empirical K-effective benchmark data utilized an uncertainty based on the standard deviation of the mean; rather than taking the standard deviation of the K-effective test population as a whole.

A FirstEnergy Nuclear Operating Company (FENOC) calculation was prepared to supplement the spent fuel pool criticality analysis of record using an alternate method to determine the bias uncertainty. The alternate method addresses the concern identified in NRC Information Notice 2011-03. Use of the alternate method resulted in K-effective values that are closer to, but remain bounded by, the K-effective limit of 0.95. Therefore, the change is not a departure from a method described in the Updated Final Safety Analysis Report and prior NRC approval of the change was not required.

<u>Change Title</u>

Hot Zero Power Steam-line Break Analysis, Revised for 9.4 Percent Power Uprate and Replacement Steam Generators

<u>Description</u>

The analysis of the hot zero power main steam line break was revised due to a difference between the calculated return to power using the transient analysis code LOFTRAN and the fuel design code ANC. This necessitated an adjustment (increase) to the input moderator coefficient in the LOFTRAN runs as required by reload safety evaluation procedures.

The hot zero power main steam line break analysis incorporates input changes, which have the effect of increasing the return to power transient (power level increases approximately 3 percent) following the event. The analysis results show that all Updated Final Safety Analysis Report acceptance criteria continue to be met. There is no damage to the fuel and departure from nucleate boiling ratio limits are met. The energy release to the containment does not cause failure of the containment structure. The resulting doses do not exceed regulatory limits. The changes do not increase the probability of an accident or malfunction. The consequences of the main steam line break (dose) are not impacted by these changes. The changes do not introduce the possibility of a new accident or malfunction. The change does not depart from a method of evaluation as described in the Updated Final Safety Analysis Report and no fission product barriers are challenged. Therefore, prior NRC approval of the change was not required.