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L-08-340

10 CFR 50.46(a)(3)(ii)

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
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
10 CFR 50.46 Report of Changes or Errors in ECCS Evaluation Models

Pursuant to 10 CFR 50.46(a)(3)(ii), FirstEnergy Nuclear Operating Company (FENOC) provides the attached report as annual notification of changes or errors in emergency core cooling system (ECCS) evaluation models for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2. Current information for both large and small break transients is provided to satisfy reporting requirements. The following attachments provide information as requested by 10 CFR 50.46:

Attachment 1 Provides a listing of each change or error in an acceptable evaluation model that affects the peak cladding temperature (PCT) calculation for particular transients. It quantifies the effects of changes that have occurred since the previous report (December 4, 2007) for the specified transients and provides an index into Attachment 2.

Attachment 2 Provides a description for each model change or error.

The PCT effects, listed in Attachment 1, result in PCTs for the large and small break loss of coolant accident (LOCA) transients as follows:

BVPS-1 Large Break LOCA - 2014°F
BVPS-1 Small Break LOCA - 1895°F
BVPS-2 Large Break LOCA - 2017°F
BVPS-2 Small Break LOCA - 1917°F

Changes or errors reflected in the PCT values above include those previously reviewed and approved by the NRC via license amendments associated with extended power uprate (TAC Nos. MC4645 and MC4646), containment conversion (TAC Nos. MC3394

A002
NRR

and MC3395), and best-estimate loss of coolant accident methodologies (TAC Nos. MC4647 and MC4648) for BVPS Unit Nos. 1 and 2 as well as those described in previous 10 CFR 50.46 reports provided through December 4, 2007.

FENOC previously committed to performing and submitting a re-analysis of the large break LOCA for BVPS Unit No. 1 within two fuel cycles following implementation of containment conversion (spring 2009) because analysis input changes resulted in PCT impacts of greater than 50 degrees Fahrenheit. This schedule has not changed.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at 330-761-6071.

Sincerely,

Peter P. Sena III

Attachments:
1. Summary of PCT Effects for BVPS LOCA Transients
2. Descriptions of Model Changes or Errors

cc: NRC Region I Administrator
NRC Senior Resident Inspector
NRR Project Manager
Director BRP/DEP
Nuclear Safety Specialist BRP/DEP

SUMMARY OF PCT EFFECTS FOR BVPS LOCA TRANSIENTS

DESCRIPTION	PCT EFFECT (°F)	ATTACHMENT 2 PAGE
BVPS-1 LARGE BREAK LOCA		
COUNTER-CURRENT FLOW LIMIT (CCFL) GLOBAL VOLUME ERROR	0	1
HOTSPOT BURST TEMPERATURE LOGIC ERRORS	0	2
BVPS-1 SMALL BREAK LOCA		
PUMP WEIR RESISTANCE MODELING	0	3
ERRORS IN REACTOR VESSEL LOWER PLENUM SURFACE AREA CALCULATIONS	0	4
BVPS-2 LARGE BREAK LOCA		
COUNTER-CURRENT FLOW LIMIT (CCFL) GLOBAL VOLUME ERROR	0	1
HOTSPOT BURST TEMPERATURE LOGIC ERRORS	0	2
BVPS-2 SMALL BREAK LOCA		
PUMP WEIR RESISTANCE MODELING	0	3
ERRORS IN REACTOR VESSEL LOWER PLENUM SURFACE AREA CALCULATIONS	0	4

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ATTACHMENT 2

DESCRIPTIONS OF MODEL CHANGES OR ERRORS

COUNTER-CURRENT FLOW LIMIT (CCFL) GLOBAL VOLUME ERROR

Background

An error was identified during the course of a Best Estimate Large Break LOCA analysis in which the volume between the core barrel and the baffle plates in the CCFL region above the active fuel length was modeled incorrectly. The corrected values have been evaluated for impact on the current licensing basis analysis results.

Affected Evaluation Models

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The CCFL global volume modeling error has been generically evaluated to have a negligible impact on PCT for affected analyses and a penalty of 0°F is assigned.

HOTSPOT BURST TEMPERATURE LOGIC ERRORS

Background

The HOTSPOT code has been updated to incorporate the following corrections to the burst temperature logic: (1) change the rod internal pressure used to calculate the cladding engineering hoop stress from the value in the previous time step to the value in the current time step; (2) revise the average cladding heatup rate calculation to reset selected variables to zero at the beginning of each trial and use the instantaneous heat-up rate when fewer than five values are available; and, (3) reflect the assumed saturation of ramp rate effects above 28°C/s for Zircaloy-4 cladding from Equation 7-66 of Reference 1.

Affected Evaluation Models

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

Sample calculations for each change showed no effect on peak cladding temperature, leading to an estimated impact of 0°F for 10 CFR 50.46 reporting purposes.

Reference 1 - WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2-5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," S. M. Bajorek et al., March 1998.

PUMP WEIR RESISTANCE MODELING

Background

Review of the reactor coolant pump data collections identified instances of either including a weir resistance for a design without a weir or double-counting the weir resistance for a design with a weir. The corrected resistances have been evaluated for impact on existing analysis results and will be incorporated into the plant-specific input databases on a forward-fit basis.

Affected Evaluation Models

1981 Westinghouse Large Break LOCA Evaluation Model with BASH
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

Resolving the identified discrepancies has been evaluated as having a negligible effect on existing results, leading to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.

ERRORS IN REACTOR VESSEL LOWER PLENUM SURFACE AREA CALCULATIONS

Background

Two errors were discovered in the calculations of reactor vessel lower plenum surface area. The corrected values have been evaluated for impact on current licensing-basis analysis results and will be incorporated on a forward-fit basis.

Affected Evaluation Models

1981 Westinghouse Large Break LOCA Evaluation Model with BASH
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

The differences in vessel lower plenum surface area are relatively minor and would be expected to produce a negligible effect on large and small break LOCA analysis results, leading to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.