FENOC™
FirstEnergy Nuclear Operating Company

5501 North State Route 2
Oak Harbor, Ohio 43449

Raymond A. Lieb
Vice President, Nuclear

419-321-7676
Fax: 419-321-7582

May 28, 2013
L-13-154

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

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

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License No. NPF-3
10 CFR 50.46 Report of Changes to or Errors in Emergency Core Cooling System Evaluation Models

In accordance with Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.46(a)(3), FirstEnergy Nuclear Operating Company (FENOC) hereby submits the annual report of changes to or errors in an emergency core cooling system (ECCS) evaluation model, or in the application of such a model, for the Davis-Besse Nuclear Power Station, Unit No. 1. The attached report covers the period from January 1, 2012 to December 31, 2012, and includes the significant changes or errors in the loss of coolant accident evaluation model that were communicated to the Nuclear Regulatory Commission (NRC) by FENOC letter dated March 16, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12076A237), supplemented by letter dated December 18, 2012 (ADAMS Accession No. ML12353A601), and determined acceptable by NRC staff by letter dated March 28, 2013 (ADAMS Accession No. ML13086A165).

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas Lentz, Manager – Fleet Licensing at (330) 315-6810.

Sincerely,

Raymond A. Lieb

Attachment:
Annual Report of Changes to or Errors in the 10 CFR 50.46 Emergency Core Cooling System Evaluation Model, or in the Application of Such a Model, for the Davis-Besse Nuclear Power Station, Unit No. 1

cc: NRC Region III Administrator
Nuclear Reactor Regulation Project Manager
NRC Resident Inspector
Utility Radiological Safety Board

Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.46(a)(3) states that each holder of an operating license shall report to the Nuclear Regulatory Commission (NRC), at least annually, each change to or error in an acceptable emergency core cooling system (ECCS) evaluation model (EM), or in the application of such a model, that affects the calculation of peak cladding temperature (PCT). The nature of the change or error and its estimated effect on the limiting ECCS analysis is to be included in the report.

In 2012, two significant changes or errors in the loss of coolant accident evaluation model were communicated to the NRC by FENOC letter dated March 16, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12076A237) and supplemented by letter dated December 18, 2012 (ADAMS Accession No. ML12353A601); thereby satisfying the 30-day reporting requirement for significant changes or errors. These changes or errors were evaluated and determined to be acceptable by NRC staff by letter dated March 28, 2013 (ADAMS Accession No. ML13086A165).

The same changes or errors are included below to satisfy the annual reporting requirement.

## EVALUATION MODEL ERRORS AND CHANGES

Babcock & Wilcox Nuclear Technologies (BWNT) Loss of Coolant Accident (LOCA) EM

1. ECCS Bypass Calculation Error

   While performing a large break loss of coolant accident (LBLOCA) sensitivity study for a 205 fuel assembly (FA) plant, a mathematical error was discovered in the RELAP5/MOD2-B&W blowdown model control variables that calculate the time for total end of bypass. AREVA NP Inc. (AREVA) identified in the BWNT LOCA EM (BAW-10192P-A, "BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," Revision 0) that the end of bypass calculations determine when an 80 percent condensation efficiency on the core flood tank injected liquid could condense all the steam reaching the upper downcomer region. The control variables incorrectly calculated the steam energy flowing into the upper downcomer region. When the control variables were corrected, the end of bypass time was predicted approximately 2 seconds earlier, resulting in a shorter lower plenum refill period with a quicker onset of lower core quench and lower PCT for the 205 FA plant limiting LBLOCA analyses. In evaluating the extent of condition for this error, a similar error was identified in all 177 FA plant LBLOCA analyses as well.

The effects of the ECCS bypass error were evaluated for the Davis-Besse 177 FA raised-loop plant configuration by utilizing a specific axial peak power location of 9.536 feet at beginning of life. The RELAP5 blowdown analysis used corrected bypass control variables for a 177-FA raised-loop plant. The ECCS bypass did not change the fuel pin heat removal in the RELAP5 blowdown analysis, but it did change the inputs to REFLOD3B, which reduced the end of bypass time by approximately 2 seconds. The previous REFLOD3B case was reanalyzed with this earlier end of bypass, and it shortened the adiabatic heatup period by 1.9 seconds. The bottom of core recovery timing and the new flooding rate from this case was provided as a boundary condition to a new BEACH analysis, and it predicted a 51.7°F decrease in the limiting PCT, which was for the unruptured node. The peak ruptured segment temperature, which is not limiting, decreased by 96.7°F. This LBLOCA PCT reduction is slightly larger than what was observed for the 177 FA lowered-loop plant core inlet peaking, and it confirms that the generic reduction of 80°F assigned to the ruptured cladding segments and the reduction of 40°F assigned to limiting unruptured cladding segments are conservative for the core exit power peaks and the raised-loop plant design.

ECCS bypass is not used for small break loss of coolant accident (SBLOCA) analyses. As a result, those analyses are not affected by this error.

2. Upper Plenum Column Weldment Modeling

During the assessment of the ECCS bypass error, another LBLOCA sensitivity study was being performed for the 205 FA plant with a revised upper plenum and upper head modeling that considered the changes in core cooling when upper plenum column weldments are explicitly modeled. This revised modeling reflects a more detailed noding arrangement in the reactor vessel upper plenum than was used and approved for application in the BWNT LOCA EM (BAW-10192P-A, Revision 0).

The modeling of the column weldment in a 177 FA raised-loop plant resulted in a lower end-of-blowdown peak fuel temperature prediction by RELAP5, but the location of the peak fuel temperature also changed. However, the unruptured segment PCT increased by 8.9°F for the raised-loop plant specific BEACH calculation. The generic column weldment estimate of a 40°F increase in the unruptured PCT is confirmed by the analysis to be bounding, and this result is consistent with the previous results based on the 177 FA lowered-loop core inlet peak evaluations. These RELAP5 and BEACH cases predict smaller increases than what was estimated for the unruptured PCT, so the generic penalty remains bounding. The non-limiting ruptured segment cladding temperature decreased for this analyzed case over the previous cases. A favorable shift in the timing of lock-in to film boiling and its influence on the cladding rupture conditions and timing confirmed that the generic estimate of an 80°F increase was bounding.

This modeling change was considered for SBLOCA, and it was concluded that it will not affect the limiting results because the SBLOCA is a slower evolving transient with up flows in the core hot bundles such that there is no net change from the presence of a column weldment in the upper plenum.

CRAFT2 LOCA EM

No errors were identified and no changes were implemented during the reporting period.

## EVALUATION MODEL APPLICATION ERRORS AND CHANGES

No errors were identified and no changes were implemented during the reporting period.

## Summary

A summary of the PCT at the beginning and end of the reporting period is provided in Table 1.

## Table 1
## 10 CFR 50.46 Summary for 2012

| | | | | |
|---|---|---|---|---|
| *Plant Name:* | | Davis-Besse Nucler Power Station, Unit No. 1 | *LOCA Spectrum* | |
| *Utility Name:* | | FirstEnergy | *Mk-B-HTP LBLOCA Full-Core* | *SBLOCA* |
| | | | *PCT or PCT Change (Δ)* | |
| *Licensing Basis at Beginning of 2012* | | | 2,119°F Estimated EM R0.10[1] | 1,780°F Estimated EM R0.10[1] |
| *2012 Licensing Activity* | | | | |
| *Item #* | *Reporting Category* | *Description* | *PCT or PCT Change (Δ)* | |
| 1 | EM Error | Emergency Core Cooling System Bypass Calculation | Δ = -80°F | Δ = 0°F |
| 2 | EM Change | Reactor Vessel Upper Plenum Column Weldment Modeling | Δ = +80°F | Δ = 0°F |
| *Licensing Basis at End of 2012* | | | 2,119°F Estimated EM R0.10[1] | 1,780°F Estimated EM R0.10[1] |

Footnote:
(1) Evaluation Model (EM) revision number.