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

July 1, 2010

Mr. Paul A. Harden
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 - ISSUANCE OF AMENDMENT REGARDING AUTOMATED STATISTICAL TREATMENT OF UNCERTAINTY METHOD IMPLEMENTATION FOR LARGE-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS (TAC NO. ME1776)

Dear Mr. Harden:

The Commission has issued the enclosed Amendment No. 286 to Renewed Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1 (BVPS-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 6, 2009, as supplemented by letter dated March 8, 2010.

The amendment revises TS 5.6.3, "Core Operating Limits Report," to allow use of the generically approved Westinghouse Topical Report, WCAP-16009-PA, "Realistic Large Break Loss of Coolant Accident Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method" for BVPS-1.

A copy of the related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosures:
1. Amendment No. 286 to DPR-66
2. Safety Evaluation

cc w/encls: Distribution via Listserv

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 286
License No. DPR-66

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee), dated July 6, 2009, as supplemented by letter dated March 8, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications contained in Appendix A, and changes to Appendix B and C, as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 286, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to startup following the fall 2010 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and Technical Specifications

Date of Issuance: July 1, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 286

RENEWED FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

Replace the following page of Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
5.6-2	5.6-2

(3) FENOC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) FENOC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) FENOC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

FENOC is authorized to operate the facility at a steady state reactor core power level of 2900 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 286, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Auxiliary River Water System

(Deleted by Amendment No. 8)

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

LCO 3.1.5, "Shutdown Bank Insertion Limits"

LCO 3.1.6, "Control Bank Insertion Limits"

LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)"

LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)"

LCO 3.2.3, "Axial Flux Difference (AFD)"

LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation" - Overtemperature and Overpower ΔT Allowable Value parameter values

LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

LCO 3.9.1, "Boron Concentration"

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology,"

WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT Trip Functions,"

WCAP-12945-P-A, Volumes 1 through 5, "Code Qualification Document for Best Estimate LOCA Analysis,"

(For Unit 1 only) WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM),"

WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification,"

WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,"

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report,"

WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicating Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids."

As described in reference documents listed above, when an initial assumed power level of 102% of RATED THERMAL POWER is specified in a previously approved method, 100.6% of RATED THERMAL POWER may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING AMENDMENT NO. 286 TO RENEWED FACILITY OPERATING

LICENSE NO. DPR-66

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

1.0 INTRODUCTION

By application dated July 6, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091890844), as supplemented by letter dated March 8, 2010 (ADAMS Accession No. ML100700241), FirstEnergy Nuclear Operating Company (the licensee), requested changes to the Technical Specifications (TSs) for Beaver Valley Power Station, Unit No. 1 (BVPS-1). The supplement dated March 8, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 1, 2009 (74 FR 62835).

The proposed change would revise TS 5.6.3, "Core Operating Limits Report," to allow use of the generically approved Westinghouse Topical Report, WCAP-16009-PA, "Realistic Large Break Loss of Coolant Accident [LOCA] Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method [ASTRUM]" for BVPS-1.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.46 provides the requirements for the emergency core cooling system (ECCS), requiring that the ECCS must be designed such that, when analyzed using the guidance set forth in 10 CFR 50.46, it demonstrates acceptable performance subject to the criteria contained in 10 CFR 50.46(b)(1) through (b)(5).

The NRC staff reviewed the licensee's request to implement the ASTRUM methodology to ensure the following:

(1) ASTRUM is generically NRC-approved and acceptable for implementation at BVPS-1, and

(2) The ASTRUM analysis demonstrates acceptable ECCS performance relative to the LOCA acceptance criteria 10 CFR 50.46(b)(1) through (b)(3).

Note that the NRC staff, per item (2) above, reviewed the licensee's large-break LOCA (LBLOCA) analysis for compliance with 10 CFR 50.46)(b)(1) through (b)(3). Separate analyses and operator actions are credited to demonstrate compliance with the remaining acceptance criteria, and these items are not affected by the implementation of the ASTRUM LBLOCA analysis.

3.0 TECHNICAL EVALUATION

3.1 ASTRUM IMPLEMENTATION

Westinghouse obtained generic NRC approval of its original topical report describing the best-estimate (BE)-LBLOCA methodology in 1996 for 3 and 4-loop pressurized-water reactors. This method is known as the Code Qualification Document (CQD) methodology (Reference 5). NRC approval of the methodology is documented in the NRC safety evaluation report (ADAMS Accession No. ML043100073) appended to the topical report.

In 2005, Westinghouse completed a program to revise the statistical approach used to develop the Peak Cladding Temperature (PCT) and oxidation results at the 95th percentile. This method is based on the CQD methodology and follows the steps in the Code Scaling Applicability and Uncertainty (CSAU) methodology (NUREG/CR- 5249). However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method in which the uncertainty parameters are simultaneously sampled for each case. The approved ASTRUM evaluation model is documented in WCAP-16009-P-A (Reference 3).

3.2 Summary of Licensee's Analysis and Comparison to Current Licensing Basis

The licensee's current licensing basis LOCA analysis is based on the 1996 BE-LBLOCA methodology, and was analyzed at the current licensed thermal power level of 2900 megawatts-thermal (MWt). The BVPS-1 Updated Final Safety Analysis Report (UFSAR) states that the current licensing basis LBLOCA analysis demonstrated a PCT of 2144 °F. The analysis also indicates that plant-specific analysis demonstrated that the limiting break for BVPS-1 was a split break.

The proposed ASTRUM analysis was performed for operating at an assumed power level of 2900 MWt, and showed a limiting PCT of 2161 °F for BVPS-1. The licensee stated that it and its vendor, Westinghouse Electric Company LLC, continue to have ongoing processes which ensure that LOCA analysis input values conservatively bound current operating values. The licensee also stated that the BE-LBLOCA analysis and associated model for BVPS-1 is unit specific. These statements, made by the licensee, provide assurance that the LOCA results are directly applicable to BVPS-1. The full results are tabulated below for BVPS-1.

Parameter	ASTRUM Results	10 CFR 50.46 Limits
Peak Cladding Temperature	2161 °F	< 2200 °F
Local Metal Oxidation	9.22%	< 17%
Core-Wide Oxidation	0.94%	< 1%

The licensee stated that WCAP-16009-P-A uses a double-ended guillotine break for plant-specific confirmatory analyses. However, the studies discussed in the request for additional information (RAI) responses, submitted as a part of the NRC staff generic review of WCAP-16009-P-A, indicated that, for the three-loop plant, the split break had a higher PCT (Reference 4). The NRC staff requested the licensee to address how the generic, double-ended guillotine break was confirmed to be limiting, since plant-specific analysis in the current licensing basis demonstrates that the limiting break is a split break.

The licensee provided additional information to demonstrate that the generic methodology studies, discussed in WCAP-16009-P-A, confirm that the limiting split and double-ended guillotine breaks have comparable PCTs, and that, generically, it is appropriate to sample from both types of breaks (Reference 2). In response to the NRC staff RAI, the licensee provided extensive comparisons of the current licensing basis analysis to the proposed ASTRUM analysis to confirm that the switch in break configuration is attributable to the specific, limiting model, and not to any changes regarding the analytic input conditions. This information demonstrated that the switch in break configuration was an acceptable and expected result arising from the modeling methodology.

The licensee also provided a plot that correlated break configuration to PCT for the limiting cases obtained using the current analysis of record (Reference 2). The plot showed that the current PCT-limiting case is a double-ended guillotine rupture. By contrast, the analysis described in the UFSAR indicates that the split break is limiting. Both of these analyses were performed using the same statistical methodology, which underscores the finding discussed above that a switch in limiting break geometry is an acceptable and expected result arising from the use of a statistical methodology.

3.3 NRC Staff's Findings

The NRC staff reviewed the information submitted by the licensee. The ASTRUM method has been approved by the NRC for the analysis of LBLOCAs at Westinghouse nuclear power plants with three reactor coolant loops, such as BVPS-1. The licensee's analysis demonstrates acceptable performance relative to the 10 CFR 50.46 acceptance criteria. Based on this, the NRC staff finds the licensee's request to implement ASTRUM for LBLOCA analysis is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (74 FR 62835). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Sena, P. P., FirstEnergy, Letter to U.S. Nuclear Regulatory Commission, "Application to Permit Operation with ASTRUM Best-Estimate Large Break Loss of Coolant Accident Methodology," Docket 50-334, July 6, 2009, ADAMS Accession No. ML091890844.

2. Harden, Paul A., FirstEnergy, Letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information on the ASTRUM Best-Estimate Large-Break Loss of Coolant Accident Methodology License Amendment Request," Docket 50-334, March 8, 2010, ADAMS Accession No. ML100700241.

3. Westinghouse Electric Company, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method," WCAP-16009, March 11, 2005, ADAMS Accession No. ML050910157.

4. Gresham, J. A., Westinghouse Electric Company, Letter to U.S. Nuclear Regulatory Commission, "Transmittal of Proprietary Information," LTR-NRC-04-30, May 11, 2004, ADAMS Accession No. ML041340596.

5. Bajorek, S. M., et. al., Westinghouse Electric Company, "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A, 1998, ADAMS Accession Package No. ML080630386.

Principal Contributor: B. Parks

Date: July 1, 2010

July 1, 2010

Mr. Paul A. Harden
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 - ISSUANCE OF AMENDMENT RE: AUTOMATED STATISTICAL TREATMENT OF UNCERTAINTY METHOD IMPLEMENTATION FOR LARGE-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS (TAC NO. ME1776)

Dear Mr. Harden:

The Commission has issued the enclosed Amendment No. 286 to Renewed Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1 (BVPS-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 6, 2009, as supplemented by letter dated March 8, 2010.

The amendment revises TS 5.6.3, "Core Operating Limits Report," to allow use of the generically approved Westinghouse Topical Report, WCAP-16009-PA, "Realistic Large Break Loss of Coolant Accident Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method" for BVPS-1.

A copy of the related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosures:
1. Amendment No. 286 to DPR-66
2. Safety Evaluation

cc w/encls: Distribution via Listserv

Amendment No.: ML101600408 *Input received. No substantive changes made.

OFFICE	LPLI-1/PM	LPLI-1/LA	SRXB/BC	ITSB/BC	OGC	LPLI-1/BC
NAME	NMorgan	SLittle	GCranston*	RElliot	LSubin	NSalgado
DATE	6/10/2010	6/10/2010	3/10/2010	6/16/2010	6/18/2010	7/01/2010

OFFICIAL RECORD COPY

DATED: July 1, 2010

AMENDMENT NO. 286 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-66
BEAVER VALLEY POWER STATION, UNIT NO. 1

PUBLIC
LPL1-1 R/F
RidNrrDorl
RidsNrrDorlLpl1-1
RidsNrrDorlDpr
RidsNrrDirsItsb
RidsNrrPMBeaverValley
RidsNrrLASLittle (paper copy)
RidsRgn1MailCenter
RidsOGCRp
RidsAcrsAcnw&mMailCenter
RidsNrrDssSrxb
BParks
RidsNrrDirsItsb