March 2, 2006

Mr. Mark B. Bezilla
Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
Mail Stop A-DB-3080
5501 North State Route 2
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SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: FRAMATOME MARK B-HTP FUEL DESIGN FOR CYCLE 15 (TAC NO. MC6888)

Dear Mr. Bezilla:

The Commission has issued the enclosed Amendment No. 274 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit 1. The amendment revises the technical specifications in response to your application dated May 2, 2005, as supplemented by letters dated August 28 and September 15, 2005, and January 12, January 13, February 9, and February 28, 2006.

The amendment revises the TSs Section 2.1.1, "Safety Limits – Reactor Core," and TS Section 2.2.1, "Limiting Safety Settings – Reactor Protection System Setpoints." The amendment supports the use of Framatome Mark B-HTP fuel design for Cycle 15, which is scheduled to begin following the refueling outage in March 2006.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/
Stephen P. Sands, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures:
1. Amendment No. 274 to NPF-3
2. Safety Evaluation

cc w/encls: See next page

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FIRSTENERGY NUCLEAR OPERATING COMPANY

AND

FIRSTENERGY NUCLEAR GENERATION CORP.

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 274
License No. NPF-3

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

   A. The application for amendment filed by FirstEnergy Nuclear Operating Company et al. (the licensee), dated May 2, 2005, as supplemented by letters dated August 28 and September 15, 2005, and January 12, January 13, February 9, and February 28, 2006, 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 as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 274, 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 its date of issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Mindy S. Landau, Acting Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 2, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 274

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
2-2	2-2
2-5	2-5
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 274 TO FACILITY OPERATING LICENSE NO. NPF-3

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, Commission) dated May 2, 2005, as supplemented by letters dated August 28 and September 15, 2005, and January 12, January 13, February 9, and February 28, 2006, FirstEnergy Nuclear Operating Company, et al. (the licensee), requested changes to the technical specifications (TSs) for the Davis-Besse Nuclear Power Station, Unit 1 (Davis-Besse). The supplements dated August 28 and September 15, 2005, and January 12, January 13, February 9, and February 28, 2006, provided additional information that clarified the application, did not expand the scope of the application as noticed, and did not change the NRC staff's proposed no significant hazards consideration determination as published in the *Federal Register* on May 24, 2005 (70 FR 29796).

The Framatome Mark B-HTP fuel design incorporates the high thermal performance (HTP) spacer grid design that reduces the likelihood of fuel rod defects caused by spacer grid to fuel rod fretting. The Mark B-HTP fuel design has higher pressure drop across the fuel assemblies that prompts the need to use a new departure from nucleate boiling (DNB) correlation, namely, the BHTP correlation, for the critical heat flux (CHF) calculation. The BHTP correlation that is documented in Topical Report BAW-10241P-A, "BHTP DNB Correlation Applied with LYNXT," has been approved by the NRC. The proposed TS changes reflect the loading of the Mark B-HTP fuel and the use of the BHTP DNB correlation that results in more restrictive safety limits and trip setpoint for the variable pressure-temperature reactor trip function.

The proposed changes would revise theTSs Section 2.1.1, "Safety Limits – Reactor Core," and TS Section 2.2.1, "Limiting Safety Settings – Reactor Protection System Setpoints." The amendment supports the use of Framatome Mark B-HTP fuel design for Cycle 15, which is scheduled to begin following the refueling outage in March 2006. Specifically, the proposed changes would revise:

TS Figure 2.1-1, "Reactor Core Safety Limit"

> TS Section 2.1.1 states that the combination of reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1, "Reactor Core Safety Limit." It is proposed to revise Figure 2.1-1 to reflect implementation of a new fuel design in Cycle 15.

<u>TS Table 2.2-1, "Reactor Protection Instrumentation Trip Setpoint"</u>

TS Section 2.2.1 states that the reactor protection system (RPS) instrumentation setpoints shall be set consistent with the allowable values (AV) shown on Table 2.2-1, "Reactor Protection System Instrumentation Trip Setpoints." As a result of the new fuel design proposed for Cycle 15, the licensee requests that the AV listed in Table 2.2-1 for Functional Unit 7, "RC Pressure-Temperature" (also referred to as "variable low pressure") be revised from "$\$(16.00\ T_{out}$ EF - 7957.5) psig" to "$\$(16.25\ T_{out}$ EF - 7899.0) psig."

2.0 REGULATORY EVALUATION

General Design Criterion (GDC) 10 in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR 50) specifies that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDL) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOO). Section 50.36(c)(1) specifies that TSs will include safety limits, limiting safety system settings (LSSS), and limiting control settings. Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity; and LSSS are settings for automatic protective devices related to those variables having significant safety functions.

The licensee, in Section 5.2 of Enclosure 1 of its submittal, identified regulatory requirements and criteria that are applicable to the systems that are affected by the proposed one-time TS changes. For the TSs, 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs include items in the following five specific categories related to station operation: (1) safety limits, LSSS, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs. As stated in 10 CFR 50.36(c)(2)(i), the "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

Paragraph (c)(1)(ii)(A) of 10 CFR 50.36 requires that the TS include LSSS. This paragraph specifies, among other things, "where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Accordingly, limits for instrument channels that initiate protective functions must be included in the TSs. Setpoints found to exceed TS limits are considered a malfunction of an automatic safety system. Such an occurrence could challenge the integrity of the reactor core, reactor coolant pressure boundary (RCPB), containment, and associated safety systems.

In accordance with 10 CFR 50.36(c)(1), Davis-Besse TS Figure 2.1-1 depicts the reactor core safety limit (RCSL) curve of the reactor core (RC) pressure as a function of reactor outlet temperature, and defines an acceptable operation region beyond which the RPS will trip the

reactor. The RC pressure-temperature (P-T) trip function is a part of the RPS to ensure the DNB SAFDL is not exceeded, in compliance with GDC 10. The NRC staff evaluation of the proposed changes to TS Figure 2.1-1 is to assure continued compliance with GDC 10 and 10 CFR 50.36(c)(1).

Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation," describes a method acceptable to the NRC staff for complying with the NRC regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

In addition, NRC provided clarification on the requirements of 10 CFR 50.36 in its letters to the Nuclear Energy Institute dated August 23 and September 7, 2005.

3.0 TECHNICAL EVALUATION

3.1 Setpoint-Related Technical Specification Considerations

a. Limiting Setpoint

The licensee derived the Limiting Setpoint as follows:

i. Develop a P-T curve consisting of the limiting acceptable P-T values from the safety analyses.

ii. The licensee indicates that the P-T curve will bow outward toward lower pressures and higher temperatures. Use a straight line connecting the endpoints in lieu of the P-T curve developed above. This line will yield a higher pressure at each temperature, and will therefore be conservative in terms of limiting the DNBR.

iii. The licensee has indicated that the slope of the P-T line must be no more than 16.25 psi per EF. Adjust the slope accordingly, and offset the line as necessary to make it contain the original upper endpoint of the P-T curve. For the purposes of this evaluation, the resulting line will be taken to be the effective analytical limit (EAL) for pressure, as a function of temperature. The modified P-T line will indicate higher pressure at each value of temperature, and is therefore conservative relative to both the originally-constructed line and the original P-T curve in terms of limiting DNBR.

iv. Add a constant pressure of 30.239 psi to the EAL, to accommodate the instrument hardware and process errors. The licensee has computed the hardware and process uncertainty to be 30.239 psi with 95/95 confidence. The licensee has also indicated that the temperature channel uncertainty is multiplied by the P-T line slope and combined with the pressure uncertainty to yield the 30.239 psi value. The result is the “RCS Pressure Temperature Trip” line shown on proposed Figure 2.1-1 and presented as the AV for Functional Unit 7 by means of the formula “(16.25 T_{out} EF - 7899.0) psig.”

v. Add an additional constant pressure margin to the AV line to yield the limiting setpoint line. The proposed margin is 13.5 psig, and includes allowances for drift, M&TE uncertainty, and calibration tolerance.

b. As-Found Setpoint Evaluation

i. Deviation Limit

The proposed TS require that an explicit evaluation of operability be performed if the as-found setpoint is not within the "predefined as-found acceptance criteria band." This must be accomplished before the channel can be declared OPERABLE. The acceptance criteria band for each test is established relative to the previous as-left setting. The width and application of this band is documented in the licensee's letter dated February 9, 2006.

ii. Allowable Value

The proposed TS require that the channel be declared inoperable if the as-found setpoint is non-conservative relative to the AV.

c. As-Left Setpoint Requirement

The proposed TS change requires that the setpoint be reset to a value within the as-left tolerance of the limiting setpoint, or to a value that is more conservative than the limiting setpoint. The proposed as-left tolerance is ±6 psi. This is documented in the licensee's letter dated February 9, 2006.

3.2 Reactor Core Safety Limit

The RCSL curve in TS Figure 2.1-1 is the analytical limit for the RC P-T trip function. This line represents the most severe P-T conditions that the core can operate in a steady-state condition without violating the DNB criterion of the plant, i.e., it corresponds to the hot-leg pressure - core exit temperature conditions at which the DNB Ratio (DNBR) is equal to or greater than the minimum DNBR limit predicted for the maximum possible overpower of 112 percent rated thermal power (RTP) and the RC flow of 380,000 gallons per minute (gpm) (compared to the minimum required measured flow of 389,500 gpm). The acceptable operating region in TS Figure 2.1-1 represents the operating window bounded by the RC High Pressure, RC Low Pressure, and variable RC P-T trip functions.

3.3 Revision to TS Figure 2.1-1

The current RCSL curve in TS Figure 2.1-1 was determined based on existing fuel designs placed in the RC for current operation. Because of the licensee-planned reload with the Mark B-HTP fuel for Cycle 15 that would result in a more restrictive "Acceptable Operation" region, the license amendment request would revise the current RCSL curve, creating a more restrictive "Acceptable Operation" region. The AV of the variable P-T trip function is also revised accordingly. No change is made to the RC Low Pressure and RC High Pressure trip functions.

The proposed revision to TS Figure 2.1-1 was based on a new thermal design limit (TDL) in accordance with an updated analysis in support of the proposed Mark B-HTP fuel design and the use of the BHTP CHF correlation. The Davis-Besse thermal hydraulic analysis was performed using the NRC-approved statistical core design (SCD) methodology described in the NRC-approved Topical Report BAW-10187P-A (Reference 1). In the SCD methodology, the uncertainty distributions of statistically-treated parameters were determined. To account for the

effects of uncertainties of these parameters, the SCD DNBR limit, referred to as the statistical design limit (SDL), is determined using a Monte Carlo propagation of these uncertainties in convolution with the DNBR limit of the DNB correlation. The TDL used for the determination of the RCSL curve and for safety analyses is generally sufficiently greater than the SDL to provide the necessary DNB margin to offset cycle-specific needs, such as the transition mixed-core DNB penalty.

In response to a NRC staff request for additional information (Reference 2), the licensee provided the derivations of the SDL, TDL, and the transition core penalty for Cycle 15 and subsequent cycles when the higher-resistance Mark-B HTP fuel assemblies co-exist with the existing fuel assemblies in the core. Table 1.A-1 of Reference 2 lists the uncertainties of parameters used to derive the SDL for Davis-Besse Cycle 15. With a DNBR limit of 1.132 for the BHTP correlation, the SDL for Davis Besse Cycle 15 is calculated (the SDL value is proprietary to the licensee). The licensee also provided the calculation of the transitional mixed core penalty. For Cycle 15 with 76 Mark-B HTP fuel assemblies in the core, the transition core DNB penalty will be approximately 0.18. The licensee also considered the effects of potential fuel densification and rod bow of the Mark B-HTP fuel design on the DNBR. Because the fuel densification effect has been accommodated in the core thermal-hydraulic model, there is no need for additional DNBR margin for fuel densification. The rod bow penalty is accommodated in the application of a flow area reduction factor treated statistically in the development of the SDL as described in the NRC-approved Topical Report BAW-10179P-A, Revision 5 (Reference 3). Therefore, no additional DNBR margin is needed for rod bow penalty.

In the analysis of the P-T safety limit line, the TDL (proprietary value) that provides about 25 percent margin from the SDL is used. This large margin between the TDL and SDL is sufficient to cover the mixed core penalty and other cycle-specific needs, and is therefore acceptable. The analysis for establishing the new safety limit was performed using NRC-approved methodologies and thermal-hydraulic codes, and the TDL based on the DNBR limit of 1.132 for the BHTP CHF correlation. The analysis result showed more restrictive RCSL with the limits curve slightly to the left of the existing curve. This corresponds to a reduction in the safety limit reactor outlet temperature for each corresponding RCS pressure. The “RC Pressure Temperature Trip” curve is also moved to the left, resulting in a reduction in the “Acceptable Operation” region depicted in TS Figure 2.1-1. Since the SDL is calculated based on the NRC-approved BHTP CHF correlation and SCD methodology, and since the revised RCSL curve in Figure 2.1-1 is based on the TDL having sufficient margin to the SDL, the operation within the revised Acceptable Operation region with the revised RC P-T trip line continues to assure that the SAFDLs will not be exceeded during AOOs. Therefore, GDC 10 will continue to be met, and the revised TS Figure 2.1-1 satisfies 10 CFR 50.36(c)(1) and is acceptable.

The NRC staff has reviewed the licensee's proposed TS changes concerning AVs and limiting setpoints, and the notes proposed to be added to the TS relating to those changes. The addition of the notes and other TS changes, together with the implementation of the limiting setpoint, as-found limits, and as-left limits as described above, provides reasonable assurance that plant will operate in accordance with the safety analyses and also that the operability of the instrumentation is ensured. Based on its review of the licensee's submittals, justifications, and the addition of the TS notes, the NRC staff finds that the proposed TS changes are acceptable.

Based on the above evaluation, the NRC staff concludes that it is safe to operate the plant using the proposed TS changes and AVs and, therefore, the proposed TSs and AVs meet

10 CFR 50.36. Based on this conclusion, the NRC staff further concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 amendment involves no significant hazards consideration and there has been no public comment on such finding (70 FR 29796, May 24, 2005). Accordingly, the amendment meets 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 amendment.

6.0 CONCLUSION

The NRC staff 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. BAW-10197P-A, “Statistical Core Design for B&W-Designed 177FA Plants,” B&W Fuel Company, March 1994.

2. Letter from Mark B. Bezilla, FirstEnergy Nuclear Operating Company, to US Nuclear Regulatory Commission, “Davis-Besse Nuclear Power Station, Supplemental Information Regarding License Amendment Application to Support Mark B-HTP Fuel Design for Cycle 15 (License Amendment Request (LAR) No. 05-0002; TAC No. MC6888),” Serial Number 3166, August 28, 2005.

3. BAW-10179P-A, Revision 5, “Safety Criteria and Methodology for Acceptable Cycle Reload Analysis,” Framatome ANP, December 2002.

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Date: March 2, 2006