January 11, 2006

Mr. James H. Lash
Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
P. O. Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2) - ISSUANCE OF AMENDMENTS RE: STEAM GENERATOR (SG) LEVEL ALLOWABLE VALUE SETPOINTS (TAC NOS. MC4649 AND MC4650)

Dear Mr. Lash:

The Commission has issued the enclosed Amendment No. 270 to Facility Operating License No. DPR-66 and Amendment No. 152 to Facility Operating License No. NPF-73 for BVPS-1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 5, 2004, as supplemented March 22, August 29, and October 31, 2005.

These amendments revise the BVPS-1 and 2 TSs 3/4.3.1, “Reactor Trip System Instrumentation,” and 3/4.3.2, “Engineered Safety Feature Actuation Instrumentation,” to modify SG level allowable value setpoints.

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

Sincerely,

/RA/

Timothy G. Colburn, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures: 1. Amendment No. 270 to DPR-66
2. Amendment No. 152 to NPF-73
3. Safety Evaluation

cc w/encls: See next page

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ACCESSION NO. ML053530143

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DATE	12/29/05	12//29/05	12/05/05	1/10/06	1/10/06

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FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 270
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 October 5, 2004, as supplemented March 22, August 29, and October 31, 2005, 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. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 270, 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 11, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 270

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

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

Remove	Insert
3/4 3-3	3/4 3-3
3/4 3-12	3/4 3-12
------	3/4 3-13a
3/4 3-19a	3/4 3-19a
3/4 3-31a	3/4 3-31a
3/4 3-32	3/4 3-32
B 3/4 3-1a	B 3/4 3-1a
B 3/4 3-1b	B 3/4 3-1b
B 3/4 3-1c	B 3/4 3-1c
B 3/4 3-1d	B 3/4 3-1d
B 3/4 3-1e	B 3/4 3-1e
B 3/4 3-1f	B 3/4 3-1f
B 3/4 3-1g	B 3/4 3-1g
B 3/4 3-1h	B 3/4 3-1h
B 3/4 3-1i	B 3/4 3-1i
B 3/4 3-1j	B 3/4 3-1j
B 3/4 3-1k	B 3/4 3-1k
B 3/4 3-1l	B 3/4 3-1l
B 3/4 3-1m	B 3/4 3-1m
B 3/4 3-1n	B 3/4 3-1n
B 3/4 3-1o	B 3/4 3-1o
B 3/4 3-1p	B 3/4 3-1p
------	B 3/4 3-1q

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 152
License No. NPF-73

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 October 5, 2004, as supplemented March 22, August 29, and October 31, 2005, 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-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 152, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 11, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 152

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

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

Remove	Insert
3/4 3-3	3/4 3-3
3/4 3-11	3/4 3-11
------	3/4 3-13a
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36
3/4 3-38	3/4 3-38
B 3/4 3-1a	B 3/4 3-1a
B 3/4 3-1b	B 3/4 3-1b
B 3/4 3-1c	B 3/4 3-1c
B 3/4 3-1d	B 3/4 3-1d
B 3/4 3-1e	B 3/4 3-1e
B 3/4 3-1f	B 3/4 3-1f
B 3/4 3-1g	B 3/4 3-1g
------	B 3/4 3-1h
------	B 3/4 3-1i

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 270 AND 152 TO FACILITY OPERATING

LICENSE NOS. DPR-66 AND NPF-73

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2)

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By application dated October 5, 2004, as supplemented March 22, August 29, and October 31, 2005, the FirstEnergy Nuclear Operating Company (FENOC, the licensee), requested changes to the Technical Specifications (TSs) for BVPS-1 and 2. The supplements dated March 22, August 29, and October 31, 2005, 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 November 23, 2004 (69 FR 68183).

The proposed changes would revise the BVPS-1 and 2 TSs 3/4.3.1, "Reactor Trip System Instrumentation," and 3/4.3.2, "Engineered Safety Feature Actuation Instrumentation," to modify steam generator (SG) level allowable value (AV) setpoints. Specifically, the proposed TS changes would increase the AVs of the SG water level-low-low setpoints from 14.6 percent and 16 percent to 19.6 percent and 20 percent of the narrow range (NR) instrument span for BVPS-1 and 2, respectively. These are the AVs of setpoints specified in TS Table 3.3-1 to initiate a reactor trip, and the actuation setpoints specified in TS Table 3.3-3 to start the auxiliary feedwater pumps. Also, for BVPS-2, the AV of the SG water level-high-high setpoint would increase from 81.1 percent to 92.7 percent of the NR span. This is the AV of a setpoint for actuation of the turbine trip and the feedwater system isolation specified in TS Table 3.3-3.

The proposed TS changes are necessary to address recent generic issues involving SG water level measurement uncertainty considerations associated with Westinghouse-designed SGs.

In support of its proposed TS changes, the licensee provided the results of its technical evaluation (Reference 1) and responses (References 2, 5, and 6) to the NRC staff's requests for additional information (RAIs).

2.0 REGULATORY EVALUATION

In accordance with General Design Criterion (GDC) 15, "Reactor Coolant System Design," of Appendix A, "General Design Criteria For Nuclear Power Plants," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), the acceptable pressure limits of the reactor coolant system (RCS) pressure boundary must not be exceeded during normal operation and anticipated operational occurrences (AOOs).

Also, the NRC issued Information Notice (IN) 2002-10 (Reference 3) on March 7, 2002, to alert holders of operating licenses to the potential for non-conservative setpoints of the SG water level. The IN was issued as a result of a February 9, 2002, occurrence at Diablo Canyon Power Plant, Unit No. 2, where the SG water level NR instrumentation did not respond as expected to initiate an automatic reactor trip and auxiliary feedwater (AFW) system actuation on the SG water level-low-low signal during a plant trip. This event prompted Westinghouse, the SG manufacturer, to issue various Nuclear Safety Advisory Letters (NSALs).

As discussed in NSAL-02-3 and its revision, Westinghouse attributed the water level measurement uncertainties mainly to a differential pressure (ΔP), previously unaccounted for, created by steam flow past the mid-deck plate in the moisture separator section of the SG. Westinghouse-designed SGs incorporate a mid-deck plate at the top of the primary separator assembly between the upper and lower taps used for the SG NR water level instruments. The installation of the mid-deck plate is to reduce moisture carryover. When some of the steam flows through the separator downcomer, instead of the primary separator orifice, this steam with some entrained moisture will flow upwards through the flow area in the mid-deck plate, creating a pressure differential. The mid-deck plate ΔP, which is a function of steam flow, causes the SG NR instrumentation to read higher than the actual water level, and adversely affects the SG level-low-low trip with an uncertainty bias in the non-conservative direction. Therefore, the SG water level instrumentation without accounting for this ΔP phenomenon could be non-conservative during certain transients.

NSAL-02-4 deals with uncertainties in the measurement created by the void content of the two-phase mixture above the mid-deck plate that is not reflected in the calculation. The NSAL indicated that the uncertainties may adversely affect the SG water level-high-high trip signal for actuation of the turbine trip and feedwater system isolation in the non-conservative direction.

NSAL-02-5 deals with potential inaccuracies in the initial water level assumed in the design transient analyses affected by SG water level uncertainties. The NSAL indicated that the analyses may not be bounding because the velocity head under some conditions may increase the uncertainties in the SG water level control system.

NSAL-03-09 indicates that Westinghouse has developed a program for the Westinghouse Owners Group that evaluates the effects on the SG water level control system uncertainties from various items. These items include the mid-deck plate, feedwater ring and feedwater ring supports, lower-deck plate supports, non-recoverable losses due to carryunder, decrease in subcooling due to carryunder, as well as transient conditions due to events such as the loss of normal feedwater, or a steamline break outside containment. Under the program, Westinghouse evaluated the design features of Westinghouse-designed SGs and other phenomena associated with Westinghouse SGs as they affect uncertainties in terms of the SG water level control system, and the SG water level-low-low and level-high-high reactor trip functions.

As a result of its evaluation of the information in Westinghouse NSALs, the licensee proposed to increase the AVs of the SG NR water level-low-low setpoints for reactor trip specified in BVPS-1 and 2 TS Table 3.3-1, Functional Unit 14, and for AFW actuation specified in BVPS-1 TS Table 3.3-3, Functional Unit 7.a and in BVPS-2 TS Table 3.3-3, Functional Unit 7.b. Also, for BVPS-2, the licensee proposed to increase the AV of the SG NR water level-high-high setpoint for turbine trip and feedwater system isolation specified in TS Table 3.3-3, Functional Unit 5.b. The NRC staff evaluated the licensee's proposed TS changes in accordance with the GDC 15 requirements and verified that the licensee appropriately addressed the issues discussed in the NSALs.

The following regulatory bases and guidance are also applicable to the systems discussed in the license amendment application.

Paragraph (c)(ii)(A) of 10 CFR 50.36, "Technical specifications," requires that the TSs include limiting safety system settings. This paragraph specifies, among other things, that "where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must also 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.

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 assuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

3.0 TECHNICAL EVALUATION

SG water level-low-low channels are part of the reactor protection system (RPS) and engineered safety feature actuation system (ESFAS). Two out of three SG channels in any SG are required to trip the reactor (as an RPS function) and start the AFW pumps (as an ESFAS function) for protection of the reactor core from a loss of heat sink in the event of a sustained steam and feedwater flow mismatch.

SG water level-high-high channels are part of the ESFAS. Two out of three SG channels in any SG are required to trip the turbine and isolate feedwater to the SGs. The SG level-high-high signal functions to prevent the SGs from overfilling with water and avoid overloading effects of water in the steam piping support design during an excess feedwater flow event.

The proposed TS changes would increase the AVs of the SG water level-low-low setpoints from 14.6 percent and 16 percent to 19.6 percent and 20 percent of the NR instrument span for BVPS-1 and 2, respectively. The changes are necessary for the correction of a non-conservative bias due to various factors (including SG mid-deck plate ΔP) previously unaccounted for. Also, for BVPS-2, the AV of the SG water level-high-high setpoint would increase from 81.1 percent to 92.7 percent of the NR span. The change is to increase operating margin while including correction of a non-conservative bias due to various measurement uncertainties previously unaccounted for.

In References 1 and 2, the licensee provided the calculations of AVs of the SG water level-low-low and level-high-high setpoints for BVPS-1 and 2, including the associated uncertainties. The calculation of the setpoint uncertainties comprised of process effects and instrumentation loop uncertainty. The allowance for process effects accounts for non-instrument related effects such as process pressure variation and mid-deck plate pressure loss. These process effects are treated as biases and are combined algebraically. Instrumentation loop uncertainties address the accuracies of instruments, such as transmitter and rack, which are independent and random accuracies. The instrumentation loop uncertainties are statistically combined using the square-root-of-the-sum-of-squares technique.

As indicated in the NSALs, the SG level setpoint uncertainties may be caused by several process measurement accuracy (PMA) terms that were previously unaccounted for. The licensee considered in its BVPS-1 and 2 setpoint uncertainty calculations the following PMA terms: (1) process pressure variations; (2) reference-leg temperature uncertainty; (3) fluid velocity effects; (4) downcomer subcooling effects; (5) dynamic losses; (6) intermediate deck plate ΔP; (7) feeding ΔP; and (8) mid-deck plate ΔP. In addressing the effects of the design transient conditions, the licensee identified the events that credited a particular SG level setpoint for consequence mitigation, and determined the physical effect of the transient conditions with respect to each PMA term identified in NSALs, and overall impact on the SG level setpoint uncertainty for those affected events. The licensee provided the results of its calculations on pages 27 and 28 of Reference 2 for BVPS-1 and 2, respectively. Further, the licensee evaluated the limiting events and the associated setpoint uncertainty to verify that each of the SG level setpoint AVs established in Reference 2 maintained acceptable allowable margin with respect to the effects of transient conditions.

3.1 SG Water Level Setpoints

3.1.1 Setpoint Methodology

The Westinghouse method used for the BVPS Units 1 and 2 TSs determines a performance based AV. According to the licensee's setpoint methodology, the AV is satisfied by verification that the channel "as-left" and "as-found" conditions about the nominal trip setpoint (NTS) are within the rack comparator setting accuracy (calibration tolerance).

In a letter dated March 31, 2005 (Reference 4), to the Nuclear Energy Institute, the NRC staff stated that the RCS pressure and thermal safety limits (SLs) are protected by the practice of setting the instrument trip setpoint at, or more conservative than, the calculated trip setpoint (TSP) that accounts for credible uncertainties. However, existing AV-based TSs for the RPS and ESFAS do not require that licensees control the instrument setting based TSP. To resolve this issue, the TSs should include a requirement to return the as-left instrument setting to the TSP established to protect the SLs. Therefore, the limiting safety system setting (LSSS) becomes the AV (setpoint plus calibration tolerance). Based on the above, the licensee has modified the TSs associated with the RPS and ESFAS functions to add the following notes (Reference 5):

1. If the as-found channel setpoint is conservative with respect to the allowable value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the allowable value, the channel shall be declared inoperable.

2. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint, or a value that is more conservative than the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and methodology used to determine the nominal trip setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in document incorporated by reference into the Updated Final Safety Analysis Report.

The above notes satisfy the intent of the regulations to protect the SLs by adjusting the trip setpoint to within the as-left tolerance, and follow the guidance of the March 31, 2005, letter. Therefore, the notes are acceptable. Also, the NRC staff has evaluated the licensee's setpoint methodology (Reference 6) and concludes that the methodology demonstrates that the trip setpoint and the as-left and as-found tolerances are established and held within specified limits to protect the SLs. Therefore, the methodology is acceptable.

3.1.2 SG Level-Low-Low Setpoint-BVPS-1 and 2

Following the guidance in NSAL-03-09 and its associated analyses, the licensee calculated (References 1 and 2) the SG level-low-low uncertainties based on the transient conditions of analyses for the following events: (1) loss of normal feedwater (LONF); (2) steamline break (SLB) outside containment; (3) small/intermediate feedwater line break (FLB); and (4) large FLB.

The licensee presented the calculated values of the NTS, channel statistical allowance (CSA), margin, and AV in respective pages 27 and 28 of Reference 2 for BVPS-1 and 2. The NTSs were determined by adding the CSA to the safety analysis limit (SAL) used in the safety analyses. The SALs used to determine the SG level-low-low NTSs and AVs were the same values used in the analyses of a LONF, SLB, and FLB for the SG level-low-low trip, and the results of the analyses demonstrated compliance with the GDC 15 requirements that require that the protection system be designed to trip the reactor to assure that the RCS pressure safety limits are not exceeded. Therefore, the NRC staff concluded that the SALs are acceptable for determining NTSs and AVs of the SG level-low-low setpoints.

The licensee indicated (Reference 1) that the SALs used in analyses of the SLB and FLB are 0.0 percent of the SG NR span for BVPS-1 and 2, and the respective SALs for the LONF analyses are 10 percent of the SG NR span for BVPS-1, and 0 percent of the SG NR span for BVPS-2. Based on its review of the licensee's results of the SG level uncertainty analyses (Reference 2), the NRC staff found that the calculated NTSs for the SG level-low-low of 20.1

percent for BVPS-1, and 20.5 percent for BVPS-2 are conservative (with margin greater than or equal to 0.3 percent). Therefore, the NRC staff concluded that the calculated NTSs adequately reflect SG water level uncertainties and are acceptable.

The TS AV is the limiting value that the trip setpoint can have when tested periodically, beyond which the instrument channel is declared inoperable and corrective action must be taken. The AVs are determined by subtracting the allowance for (1) instrument calibration uncertainties, (2) instrument uncertainties during normal operation, and (3) instrument drift from the NTS. The licensee did not apply all of the terms allowed, resulting in conservative AVs. The methodology for calculating the AVs is described (Reference 2) in the previously approved reports, WCAP-11419 Revision 2, “Westinghouse Setpoint Methodology for Protection Systems Beaver Valley Power Station - Unit 1" dated December 2000, and WCAP-11366 Revision 4, “Westinghouse Setpoint Methodology for Protection Systems Beaver Valley Power Station - Unit 2" dated December 2000. This methodology credits only a rack drift uncertainty in calculating AVs and is conservative. Therefore, the proposed AVs of greater than or equal to 19.6 percent for BVPS-1 and 20.0 percent for BVPS-2 are acceptable, as the proposed AVs were determined based on acceptable methodologies. In addition, inclusion of the notes described in Section 3.1.1 of this safety evaluation (SE) to the TS instrumentation provides reasonable assurance that the plant will operate in accordance with the safety analyses and the operability of the instrumentation is assured. Based on its review of the licensee’s setpoint calculations and justifications, the NRC staff finds that the proposed TS changes to the AVs are acceptable.

3.1.3 SG Level-High-High Setpoint - Unit 2

The SG level-high-high trip is used in the design basis analyses to terminate an excessive feedwater flow event. The licensee proposed to increase the SG level-high-high NTS and AV for BVPS-2 in order to increase operating margin while including measurement uncertainties.

NSAL-02-4 indicated that the void content of the two-phase mixture above the mid-deck plate was not reflected in the SG level setpoint calculations. This results in an actuation of the SG water level-high trip signal for the turbine trip and feedwater system isolation in a non-conservative direction. In addressing the NSAL-02-4 issue, the licensee calculated the total channel uncertainty including the effects of the void content of the two-phase above the mid-deck plate. The licensee presented the calculated values of NTS, CSA, and AV for the SG level-high-high trip setpoint on page 28 of Reference 1 for BVPS-2. The NTS was determined by subtracting the CSA from the SAL .

The SAL of 96.7 percent span used to determine the SG level-high-high NTS considered the effect of the void content and was bounded by that used in the safety analyses for the SG level-high-high trip. In addition, the proposed SG level-high-high NTS of 92.2 percent span for BVPS-2 is less than that calculated by subtracting the CSA from the applicable SAL. This lower SG level-high-high NTS results in an earlier turbine trip and feedwater isolation, and thus provides a greater margin to protect the SG from overfilling with water during an excessive feedwater flow event. Therefore, the proposed NTS is conservative and acceptable.

The methodology for calculating the AV is described (Reference 2) in the previously approved report, WCAP-11366, Revision 4 for BVPS-2. This methodology credits only a rack drift (RD) uncertainty in calculating the AV. The proposed AV of less than or equal to 92.7 percent span for BVPS-2 is acceptable, as the proposed AV was determined based on an acceptable methodology. In addition, inclusion of the notes described in Section 3.1.1 of this SE to the TS instrumentation provides reasonable assurance that the plant will operate in accordance with the safety analyses and the operability of the instrumentation is assured. Based on its review of the licensee's setpoint calculations and justifications, the NRC staff finds that the proposed TS changes to the AVs are acceptable.

3.2 Initial SG Water Inventory Assumed in the Design Basis Analysis

NSAL-02-5 identified potential inaccuracies in the initial water level assumed in the safety analyses affected by SG water level uncertainties. The safety analyses may not be bounding because the velocity head under some conditions may increase the uncertainties in the SG water level control system.

In addressing the NSAL-02-5 issue, the licensee indicated (Reference 1) that the SG level control system for BVPS-1 and 2 is programmed with a reference level of 44 percent span at full-power conditions. The instrumentation uncertainties were included in modeling the initial SG water level for design-basis analyses. The SG water level was assumed in the most conservative direction for each case. For example, in the FLB analysis, it is conservative to (1) assume a high initial SG water level in the faulted SG because it delays the time of reactor trip caused by an SG water level-low-low trip signal, and (2) assume a low initial SG water level in the intact SG because it minimizes the SG mass available for post trip heat removal. Both effects of a delayed reactor trip and minimum SG inventory for heat removal will result in a greatest peak pressurizer pressure. The licensee performed uncertainty calculations for the BVPS-1 and 2 SG water level and confirmed that the calculated values of SG level uncertainties were bounded by those assumed in the initial SG water level for the analyses of record. Therefore, the NRC staff agreed with the licensee that the uncertainties discussed in NSAL-02-5 do not adversely affect the current safety analyses of record.

3.3 Anticipated Transients Without Scram (ATWS) SG Level -Low-Low Setpoint

An ATWS event is defined as an AOO combined with an assumed failure of the reactor trip system to shutdown the reactor. For Westinghouse plants, the ATWS rule, 10 CFR 50.62, requires the installation of an ATWS mitigation system actuation circuitry (AMSAC) to initiate turbine trip and actuate AFW independent of the reactor protection system. The licensee has met the ATWS rule by installing an AMSAC. The licensee indicated that actuation of the AMSAC system at BVPS-1 and 2 is initiated on a low feedwater flow condition. Since the AMSAC system is not actuated on an SG level-low-low condition, the licensee stated and the NRC staff agreed that the SG water level uncertainty issues do not affect the operation of the AMSAC system for BVPS-1 and 2.

3.4 Conclusion

The NRC staff has reviewed the licensee's proposed changes to BVPS-1 and 2 TSs to increase the AVs of the SG level-low-low trip setpoint from 14.6 percent to 19.6 percent of the narrow range instrument span for BVPS-1, and from 16 percent to 20 percent span for BVPS-2 in TS Tables 3.3-1 and 3.3-3. The AV of the SG water level-high-high setpoint for BVPS-2 also increases from 81.1 percent to 92.7 percent span in TS Table 3.3-3. Based on the evaluation discussed above, the NRC staff concludes that the licensee appropriately addressed the issues specified in the NSALs, and the proposed AVs are conservative and comply with the GDC 15 requirements. Therefore, the NRC staff concludes that the licensee's proposed TS changes are 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 and change surveillance requirements. 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 (69 FR 68183). 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. Letter from L. W. Pearce (FENOC) to USNRC, "Beaver Valley Power Station, Unit Nos. 1 and 2, Docket Nos. 50-334 and 50-412, License Nos. DPR-66 and NPF-73, License Amendment Request (LARs) Nos. 327 and 197 on Steam Generator Level Allowable Value Setpoints," dated October 5, 2004.

2. Letter from L. W. Pearce (FENOC) to USNRC, "Beaver Valley Power Station, Unit Nos. 1 and 2, Docket Nos. 50-334 and 50-412, License Nos. DPR-66 and NPF-73, Response to Request for Additional Information in Support of LAR Nos. 327 and 197, Steam Generator Level Allowable Value Setpoints," dated March 22, 2005.

3. NRC Information Notice 2002-10, "Nonconservative Water Level Setpoint on Steam Generators," dated March 7, 2002.

4. Letter from J. A. Lyons (NRC) to A. Marion (NEI), "Instrumentation, Systems, and Automation Society S67.04 Methods for Determining Trip Setpoints and Allowable Values for Safety-Related Instrumentation," dated March 31, 2005.

5. Letter from T. S. Cosgrove (FENOC) to USNRC, "Beaver Valley Power Station, Unit Nos. 1 and 2, BV-1, Docket Nos. 50-334 and 50-412, License Nos. DPR-66 and NPF-73, Supplement to License Amendment Nos. 327/197 (Unit No. 1 TAC No. MC4649/Unit No. 2 TAC No. MC4650), 317/190 (Unit No. 1 TAC No. MC3394/Unit No. 2 TAC No. MC3395) and 320 (Unit No. 1 TAC No. MC6725)," dated October 31, 2005.

6. Letter from L. W. Pearce (FENOC) to USNRC, "Beaver Valley Power Station, Unit Nos. 1 and 2, Docket Nos. 50-334 and 50-412, License Nos. DPR-66 and NPF-73, Response to Request for Additional Information in Support of LAR Nos. 327 and 197, Steam Generator Level Allowable Value Setpoints," dated August 29, 2005.

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