Mr. L. W. Myers March 22, 2001
Senior Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: DESIGN-BASIS ACCIDENT DOSE CONSEQUENCE CALCULATION REVISIONS (TAC NOS. MA9059 AND MA9060)

Dear Mr. Myers:

The Commission has issued the enclosed Amendment No. 237 to Facility Operating License No. DPR-66 and Amendment No. 119 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2). These amendments consist of changes to the Updated Final Safety Analysis Reports (UFSARs) in response to your application dated May 12, 2000, as supplemented by letters dated June 19, November 2, and December 1, 2000, and January 29, 2001.

These amendments approve changes to various design-basis accident (DBA) dose consequence calculations. For BVPS-1, changes involve the following DBAs: loss of offsite alternating-current (AC) power, fuel-handling accident, accidental release of waste gas, steam generator tube rupture, rod cluster control assembly ejection, single reactor coolant pump locked rotor, and loss of reactor coolant for small ruptured pipes/loss-of-coolant accidents. For BVPS-2, changes involve the following DBAs: steam system piping failures (or main steam line break), loss of AC power, reactor coolant pump shaft seizure, rod cluster control assembly ejection, failure of small lines carrying primary coolant outside containment, steam generator tube rupture, loss-of-coolant accidents, and waste gas system failure.

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/

Lawrence J. Burkhart, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures: 1. Amendment No. 237 to DPR-66
2. Amendment No. 119 to NPF-73
3. Safety Evaluation

cc w/encls: See next page

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Mr. L. W. Myers March 22, 2001
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These amendments approve changes to various design-basis accident (DBA) dose consequence calculations. For BVPS-1, changes involve the following DBAs: loss of offsite alternating-current (AC) power, fuel-handling accident, accidental release of waste gas, steam generator tube rupture, rod cluster control assembly ejection, single reactor coolant pump locked rotor, and loss of reactor coolant for small ruptured pipes/loss-of-coolant accidents. For BVPS-2, changes involve the following DBAs: steam system piping failures (or main steam line break), loss of AC power, reactor coolant pump shaft seizure, rod cluster control assembly ejection, failure of small lines carrying primary coolant outside containment, steam generator tube rupture, loss-of-coolant accidents, and waste gas system failure.

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Sincerely,
/RA/
Lawrence J. Burkhart, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures: Safety Evaluation
Enclosures: 1. Amendment No. 237 to DPR-66
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cc w/encls: See next page

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PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 119
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 May 12, 2000, as supplemented on June 19, November 2, and December 1, 2000, and January 29, 2001, 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, changes to the Updated Final Safety Analysis Report (UFSAR) to reflect the revisions made to selected design-basis accident dose consequence analyses, as described in the attached safety evaluation, and as set forth in the application for amendment dated May 12, 2000, as supplemented June 19, November 2, and December 1, 2000, and January 29, 2001, are authorized.

3. This license amendment is effective as of the date of its issuance and shall be implemented by the next update to the UFSAR as required by 10 CFR 50.71(e). Implementation of the amendment requires the incorporation in the UFSAR of the changes to the description of the facility as described in the licensee's application dated May 12, 2000, as supplemented June 19, November 2, and December 1, 2000, and January 29, 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/ P Tam for

Marsha Gamberoni, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: March 22, 2001

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 119
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 May 12, 2000, as supplemented on June 19, November 2, and December 1, 2000, and January 29, 2001, 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, changes to the Updated Final Safety Analysis Report (UFSAR) to reflect the revisions made to selected design-basis accident dose consequence analyses, as described in the attached safety evaluation, and as set forth in the application for amendment dated May 12, 2000, as supplemented June 19, November 2, and December 1, 2000, and January 29, 2001, are authorized.

3. This license amendment is effective as of the date of its issuance and shall be implemented by the next update to the UFSAR as required by 10 CFR 50.71(e). Implementation of the amendment requires the incorporation in the UFSAR of the changes to the description of the facility as described in the licensee's application dated May 12, 2000, as supplemented June 19, November 2, and December 1, 2000, and January 29, 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/ P. Tam for

Marsha Gamberoni, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: March 22, 2001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 237 AND 119 TO FACILITY OPERATING

LICENSE NOS. DPR-66 AND NPF-73

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By letter dated May 12, 2000, as supplemented by letters dated June 19, November 2, and December 1, 2000, and January 29, 2001, the FirstEnergy Nuclear Operating Company, et al. (FENOC, the licensee), submitted a request for changes to the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2), Updated Final Safety Analysis Reports (UFSARs). The requested changes would revise the BVPS-1 and 2 calculated doses and associated descriptions and information listed in the UFSARs for selected design-basis accidents (DBAs). For BVPS-1, the dose consequences are revised for the following DBAs: loss of offsite alternating-current (AC) power, fuel-handling accident (FHA), accidental release of waste gas, steam generator tube rupture (SGTR), rod cluster control assembly ejection, single reactor coolant pump locked rotor, and loss of reactor coolant for small ruptured pipes/loss-of-coolant accidents (LOCAs). For BVPS-2, the dose consequences are revised for the following DBAs: steam system piping failures or main steam line break (MSLB), loss of AC power, reactor coolant pump shaft seizure, rod cluster control assembly ejection, failure of small lines carrying primary coolant outside containment, SGTR, LOCAs, and waste gas system failure.

The June 19, 2000, letter revised the licensee’s no significant hazards evaluation and was used, with the original submittal, as a basis for the Nuclear Regulatory Commission (NRC) staff’s initial proposed no significant hazards consideration determination which was published on September 6, 2000 (65 FR 54086). The January 29, 2001, letter clarified that the dose calculations regarding the BVPS-1 postulated MSLB accident submitted via letter dated July 21, 2000 (FENOC License Amendment Request No. 284), superseded those forwarded in the May 12, 2000, amendment request (FENOC License Amendment Request No. 280). The

November 2, and December 1, 2000, and January 29, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The NRC staff has reviewed the licensee's proposed changes to the UFSAR for each BVPS unit with regard to the radiological consequences resulting from the postulated DBAs. The calculated doses resulting from these revisions are reported in Tables 1-1 and 1-2 for BVPS-1, and Tables 2-1 and 2-2 for BVPS-2 (Attachments 1 and 2).

2.0 EVALUATION

2.1 Revisions Common to Both Units

The licensee recalculated reactor core inventory for each BVPS unit using updated fuel and operating parameters for that unit. This, as well as other parameter updates, supported recalculation of activity concentration for the primary coolant, secondary coolant, secondary steam design and TS limits. This work was done using the methodology of NUREG/CR-0200, and ORIGEN and the SWEC proprietary ACTIVITY computer codes. The NRC staff finds the recalculated core inventory and coolant activity concentrations for BVPS-1 and 2 acceptable for use in design-basis dose analyses.

Revised atmospheric dispersion factors were reviewed and accepted by the NRC staff for BVPS-1 in License Amendment No. 205 dated September 16, 1997, (available in the NRC Public Document Room (PDR) under Accession No. 9709240310), and for BVPS-2 in License Amendment Nos. 101 dated August 18, 1999 (available in the NRC PDR under Accession No. 99083230019), and 103 dated November 18, 1999 (accessible electronically from the ADAMS Public Library component on the NRC website, http:\\www.nrc.gov (the electronic reading room) under Accession No. ML993320048). The licensee had previously implemented the revised atmospheric dispersion factors for only a subset of the design-basis dose analyses, but used the revised factors for all design-basis dose analyses performed in support of the subject amendment request. The NRC staff finds the licensee's use of the revised atmospheric dispersion factors acceptable. The licensee also performed all the revised dose analyses with thyroid dose conversion factors taken from the *International Commission on Radiation Protection (ICRP) Publication -30.* The NRC staff finds the use of ICRP-30 dose conversion factors acceptable.

2.2 Changes to Parameter Values Used in DBA Dose Consequence Analyses

Various plant parameters used in the licensee's reevaluation of dose calculations deviate from the values previously applied in the existing dose calculations. The licensee has provided the justifications for these changes and the NRC staff's evaluation is discussed below.

2.2.1 Power Level

The current value of 2766 Mega-watts thermal (MWt) has been revised to 2705 MWt for BVPS-1 and 2 for all revised DBA dose consequence analyses discussed in this safety evaluation. This change was made to reflect the current licensed maximum reactor power level in the safety analyses. BVPS-1 and 2 are limited to 2652 MWt by the facility operating license and as defined in the technical specification (TS) definition for rated thermal power. The safety

analyses are currently performed assuming 102 percent of full-power operation or 2705 MWt. The NRC staff finds that the change to the power level input parameter accurately reflects the actual licensed maximum power level and includes an appropriate margin to account for uncertainties and is, therefore, acceptable.

2.2.2 Reactor Coolant System (RCS) and Steam Generator (S/G) Fluid Content

In the BVPS-1 UFSAR, the values of RCS fluid content have been reduced in the licensee's reevaluation of dose calculations for various DBAs. The licensee states in its letter dated December 1, 2000, that the revised values are derived from WCAP-13707-1 for the 30-percent S/G tube plugging limits and they are selected as the most conservative inputs for dose calculations. These revised RCS fluid content values do not include the pressurizer vapor space. These values correspond to full power operation and result in the most limiting radiological consequences. For accidents where a coincident iodine spike is part of the source term, mixing in the pressurizer liquid volume is not assumed for conservatism, and a lower RCS volume is used for this portion of the dose calculation. The revised values of the S/G liquid content are higher and the S/G steam content are lower than the current values. These revised values were developed by Westinghouse (W) based on sensitivity studies to assure these parameters will result in conservative dose calculations for various accident scenarios.

In the BVPS-2 UFSAR, the revised value of RCS fluid content is lower than the current value since the pressurizer vapor space has been deducted from the current value of RCS liquid content. Similar to that in BVPS-1, the revised values of S/G liquid content are higher and the S/G steam content are lower than the current values. These revised values are derived from WCAP-13798-0 for the S/G tube plugging limits, power level and uncertainty as provided by W. Sensitivity studies have been performed to assure that these revised values would result in the most conservative dose calculations for various accident scenarios. The NRC staff has reviewed these revised parameters and find them acceptable due to the conservative nature of the changes.

2.2.3 Primary-to-Secondary Leak Rates

The current primary-to-secondary leak rate values of 500 gallons-per-day (gpd) in any one S/G and 1 gallon-per-minute (gpm) in all 3 S/Gs have been revised to 150 gpd and 450 gpd, respectively. These revised values reflect the allowable current leak rates provided in the TSs for both BVPS Units. Since the revised values are consistent with the current TS limits, the NRC staff finds these changes acceptable.

3.0 BVPS-1 DBA DOSE ANALYSIS REVISIONS

The NRC staff reviewed the licensee's referenced submittals with regard to the dose consequences of the revised DBA dose calculations. The NRC staff's evaluation follows. Exclusion Area Boundary (EAB) doses are calculated over the first 2 hours following the accident and all other doses are calculated over the duration of the accident.

3.1 Loss of Offsite Alternating Current (AC) Power

The changes made to this accident analysis, as documented in revisions to BVPS-1 UFSAR, Table 14.1-3, reflect corrected or conservative analysis input parameter values or input

assumptions based on plant design and operation. The analysis methodology remained the same as had been previously reviewed and approved by the NRC staff for BVPS-1, and the revised analysis resulted in no increases in any calculated doses. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as NUREG-0800, “Standard Review Plan for the Review of Safety Analyses Reports for Nuclear Power Plants” (SRP), and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC-19), the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee’s submittals, acceptable.

3.2 Fuel-Handling Accident (FHA)

The changes made to the FHA dose analysis, as documented in revisions to the BVPS-1 UFSAR, Section 14.2.1, and Tables 14.2-6, and 14.2-6a, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. The analysis methodology remained the same as had been previously reviewed and approved by the NRC staff for BVPS-1. Because the FHA dose analysis takes credit for removal of organic iodine by the supplemental leak collection and release system (SLCRS), the licensee added a safety factor of $\geq$ 2 in accordance with guidance given in Generic Letter (GL) 99-02. GL 99-02 guidance included testing nuclear-activated charcoal filters to a more stringent requirement (supported by the safety factor) than that assumed in the safety analysis to conservatively account for potential degradation to nuclear-grade charcoal filters over the surveillance interval. As a consequence of this safety factor, the calculated doses increased. The calculated thyroid dose at the EAB increased from 14.6 rem to 24.6 rem, while the calculated control room operator thyroid dose increased from 3.2 rem to 6.26 rem.

The calculated offsite doses remain well within the dose guideline values given in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100. As defined in SRP, Section 15.7.4, “Radiological Consequences of Fuel Handling Accidents,” the term “well within” means 25 percent or less of the 10 CFR Part 100 values for offsite dose (i.e., dose acceptance criteria of 75 rem thyroid and 6.25 rem whole body). The licensee’s calculated control room dose results for the FHA remain within the dose guidelines given in 10 CFR Part 50, Appendix A, GDC-19 as 5 rem whole body or its equivalent to any part of the body. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee’s submittals, acceptable.

3.3 Accidental Release of Waste Gas

The changes made to the dose analysis of an accidental release of waste gas, as documented in revisions to the BVPS-1 UFSAR, Section 14.2.3, and Table 14.2-8, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation as well as some changes to the analysis methodology. The changes in the analysis methodology were based on Branch Technical Position ETSB 11-5 of the SRP. The licensee determined the bounding waste gas release quantity, therefore, bounding the postulated dose consequences. As a result of the revisions to the dose analysis, the calculated control room whole-body dose increased from <0.01 rem to 0.0295 rem.

The calculated offsite doses remain a small fraction of the 10 CFR Part 100 dose guidelines and are within the 500 milli-rem (mrem) whole-body dose guideline given in Branch Technical Position ETSB 11-5 of the SRP. "A small fraction" is defined as 10 percent or less of the 10 CFR Part 100 dose guideline values. The calculated control room doses remain within the dose guidelines of GDC-19. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

3.4 Steam Generator Tube Rupture (SGTR)

The changes made to the SGTR dose analysis, as documented in revisions to the BVPS-1 UFSAR, Section 14.2.4, and Table 14.2-9, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. In addition, the methodology for the offsite dose analysis was changed to that of the current SGTR analysis of record for the control room operator dose. As a result of the revisions to the dose analysis, the calculated thyroid dose at the EAB for the coincident iodine spike increased from 0.9 rem to 1.37 rem.

The calculated offsite doses remain within a small fraction of the dose guideline values given in 10 CFR Part 100. As defined in SRP, Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure," a small fraction is 10 percent or less of the 10 CFR Part 100 values for offsite dose (i.e., dose acceptance criteria of 30 rem thyroid and 2.5 rem whole body). The licensee's calculated control room dose results for the SGTR remain within the GDC-19 dose guidelines of 5 rem whole body or its equivalent to any part of the body. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

3.5 Rod Cluster Control Assembly Ejection

The changes made to the dose analysis for this accident, as documented in revisions to the BVPS-1 UFSAR, Table 14.2-12, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. The analysis methodology remained the same as had been previously reviewed and approved by the staff for BVPS-1. There were no increases in any calculated doses as a result of the revisions.

The calculated offsite doses remain well within the dose guideline values given in 10 CFR Part 100. As defined in SRP Section 15.4.8, "Radiological Consequences of a Control Rod Ejection Accident," Appendix A, the term "well within" means 25 percent or less of the 10 CFR Part 100 values for offsite dose (i.e., dose acceptance criteria of 75 rem thyroid and 6.25 rem whole body). The calculated control room dose results remain within the GDC-19 dose guidelines of 5 rem whole body or its equivalent to any part of the body. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

3.6 Single Reactor Coolant Pump Locked Rotor

The changes made to the locked rotor accident dose analysis, as documented in revisions to the BVPS-1, UFSAR, Section 14.2.7, and Table 14.2-4b, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. In its previous analysis of record, the licensee assumed both a coincident iodine spike and 18-percent failed fuel. SRP 15.3.3 guidance encourages the use of either of the assumptions but not both. The licensee removed the assumption of the occurrence of a coincident iodine spike because assuming 18-percent failed fuel is more conservative than assuming the iodine spike occurrence. The calculated dose consequences resulting from assuming 18-percent failed fuel are more severe than the calculated dose consequences resulting from the iodine spike occurrence. The revised analysis did not result in any increase in calculated doses for this accident. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

3.7 Loss of Reactor Coolant from Small Ruptured Pipes/Loss-of-Coolant Accidents (LOCAs)

The changes made to the dose analysis for the LOCA, as documented in revisions to the BVPS-1 UFSAR, Section 14.3.5, and Tables 14.3-10, 14.3-13, and 14.3-14a, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. In addition, some analysis methodology was revised. Shine from the area beneath the control room that is not within the control room ventilation envelope was added as an additional contributor to the control room dose. Also, because the LOCA dose analysis takes credit for removal of organic iodine by the supplemental leak collection and release system (SLCRS), the licensee added a safety factor of $\geq$ 2 in accordance with guidance given in GL 99-02.

As a result of the changes to the LOCA dose analysis described above, the calculated control room whole body dose increased from 0.17 rem to 0.71 rem. The calculated control room dose results remain within the GDC-19 dose guidelines of 5 rem whole body or its equivalent to any part of the body. No increases in calculated offsite doses resulted from the revisions and the calculated offsite doses remain within the 10 CFR Part 100 dose guideline values of 25 rem whole body and 300 rem thyroid. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

3.7.1 Iodine Removal Coefficients

The modifications to this analysis included a determination of radiation doses caused by the radioactive iodine released to the containment after a LOCA. This source of radiation decreases with time when radioactive iodine is removed from the containment atmosphere. There are several different mechanisms for removing iodine. In its analysis, the licensee considered only iodine removal by the containment sprays which is the most significant iodine removal mechanism. The removal of iodine by sprays depends on several plant-specific parameters which are expressed in terms of iodine removal coefficients. Since, in the

containment atmosphere, most of the iodine exists either in elemental form as I_2 or in particulate form as CsI and each of them is governed by different removal mechanisms, there are two types of iodine removal coefficients, one for elemental and one for particulate form. In addition to the iodine removal coefficients, the licensee also specified a decontamination factor which determines the upper limit for removal of elemental iodine from the containment atmosphere.

For BVPS-1, the licensee calculated elemental iodine removal coefficients for seven time periods and for both quench and recirculation phases. All of the licensee-calculated values were conservative (lower) relative to those obtained by the NRC staff. Additional conservatism was introduced in actual calculation of dose rates because the licensee used the value of 10 hr^{-1} for elemental iodine removal coefficient, which is lower than the values obtained by calculation. The licensee performed similar calculations for particulate iodine. The licensee's resulting removal coefficient for particulate iodine was also conservative. Similarly, comparison of decontamination factors assumed by the licensee and calculated by the staff indicated the licensee's value to be lower and more conservative.

Based on the results of its verification, the NRC staff finds that the values utilized by the licensee represent conservative estimates for the iodine removal coefficients and are, therefore, acceptable.

4.0 BVPS-2 DBA DOSE ANALYSIS REVISIONS

4.1 Steam System Piping Failures

This event may also be referred to as the main steam line break (MSLB) accident. The changes made to this accident analysis, as documented in revisions to the BVPS-2 UFSAR, Section 15.1.5, and Table 15.1-3, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. The analysis methodology remained the same as had been previously reviewed and approved by the NRC staff for BVPS-2, and the revised analysis did not show an increase in any calculated doses. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

4.2 Loss of AC Power

The changes made to this accident analysis, as documented in revisions to the BVPS-2 UFSAR, Section 15.2.6, and Table 15.2-2, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. The analysis methodology remained the same as had been previously reviewed and approved by the staff for BVPS-2, and the revised analysis did not show an increase in any calculated doses. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

4.3 <u>Reactor Coolant Pump Shaft Seizure</u>

The changes made to this accident analysis, as documented in revisions to the BVPS-2 UFSAR, Section 15.3.3, and Table 15.3-3, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. Unlike the previous analysis of record, isolation of the control room was not assumed to occur for the revised analysis. This revised assumption is more conservative than the assumption previoulsy approved for the accident analysis. The control room isolation function remains operationally unchanged; it is just not credited in the analysis. Due to these changes, the calculated control room operator thyroid dose increased from 1.7 rem to 7.46 rem.

The updated offsite doses remain unchanged and within a small fraction of the dose guideline values given in 10 CFR Part 100. A small fraction is defined as 10 percent or less of the 10 CFR Part 100 values for offsite dose (i.e., dose acceptance criteria of 30 rem thyroid and 2.5 rem whole body). The licensee's calculated control room dose results for the locked rotor accident remain within the GDC-19 dose guidelines of 5 rem whole body or its equivalent to any part of the body. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

4.4 <u>Rod Cluster Control Assembly Ejection</u>

The changes made to this accident analysis, as documented in revisions to the BVPS-2 UFSAR, Section 15.4.8, and Table 15.4-3, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. The analysis methodology remained the same as had been previously reviewed and approved by the NRC staff for BVPS-2. No increases in any calculated doses resulted from the revised analysis.

The calculated offsite doses remain well within the dose guideline values given in 10 CFR Part 100. The term "well within" means 25 percent or less of the 10 CFR Part 100 values for offsite dose (i.e., dose acceptance criteria of 75 rem thyroid and 6.25 rem whole body). The calculated control room dose results remain within the GDC-19 dose guidelines of 5 rem whole body or its equivalent to any part of the body. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

4.5 <u>Failure of Small Lines Carrying Primary Coolant Outside Containment</u>

The changes made to this accident analysis, as documented in revisions to the BVPS-2 UFSAR, Section 15.6.2, and Table 15.6-2, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. The analysis methodology remained the same as had been previously reviewed and approved by the staff for BVPS-2. No increase in any calculated doses resulted from the revised analysis.

The updated offsite doses remain within a small fraction of the dose guideline values given in 10 CFR Part 100. A small fraction is defined as 10 percent or less of the 10 CFR Part 100 values for offsite dose (i.e., dose acceptance criteria of 30 rem thyroid and 2.5 rem whole body). The licensee's calculated control room dose results for the locked rotor accident remain within the dose guidelines given in GDC-19 as 5 rem whole body or its equivalent to any part of the body. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

4.6 Steam Generator Tube Rupture (SGTR)

The changes made to the SGTR accident dose analysis, as documented in revisions to the BVPS-2 UFSAR, Section 15.6.3, and Table 15.6-5b, reflect corrected or conservative analysis input parameter values or assumptions based on plant design and operation. The analysis methodology remained the same as had been previously reviewed and approved by the NRC staff for BVPS-2. No increase in any calculated doses resulted from the revised analysis.

The calculated offsite doses continue to meet dose acceptance criteria given in SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure." The licensee's calculated control room dose results for the SGTR remain within the GDC-19 dose guidelines of 5 rem whole body or its equivalent to any part of the body. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

4.7 Loss-of-Coolant Accidents (LOCAs)

The changes made to the LOCA dose analysis, as documented in revisions to the BVPS-2 UFSAR, Section 15.6.5, and Tables 15.6-11 and 15.6-12, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. The analysis methodology remained the same as had been previously reviewed and approved by the NRC staff for BVPS-2.

As a result of the revisions, the calculated control room operator whole-body dose increased from 0.32 rem to 0.33 rem and the calculated control room operator thyroid dose increased from 1.3 to 2 rem. There were no increases in calculated offsite doses and the offsite doses remain within the 10 CFR Part 100 dose guideline values of 25 rem whole body and 300 rem thyroid. The calculated control room dose results remain within the GDC-19 dose guidelines of 5 rem whole body or its equivalent to any part of the body. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee's submittals, acceptable.

4.7.1 Iodine Removal Coefficients

The modifications to this analysis included a determination of radiation doses caused by the radioactive iodine released to the containment after a LOCA. This source of radiation decreases with time when radioactive iodine is removed from the containment atmosphere. There are several different mechanisms for removing iodine. In its analysis, the licensee considered only iodine removal by the containment sprays which is the most significant iodine removal mechanism. The removal of iodine by sprays depends on several plant-specific parameters which are expressed in terms of iodine removal coefficients. Since, in the containment atmosphere, most of the iodine exists either in elemental form as I_2 or in particulate form as CsI and each of them is governed by different removal mechanisms, there are two types of iodine removal coefficients, one for elemental and one for particulate form. In addition to the iodine removal coefficients, the licensee also specified a decontamination factor which determines the upper limit for removal of elemental iodine from the containment atmosphere.

For BVPS-2, the licensee calculated iodine removal coefficients which were not time-dependent. There was, therefore, only one coefficient for the elemental and one for particulate iodine removal. When compared to the iodine removal coefficients calculated by the NRC staff, both of the licensee-utilized coefficients are conservative and lower. Also, the value of the decontamination factor assumed by the licensee for BVPS-2, was lower than the value calculated by the NRC staff. Consequently, the iodine removal coefficients are more conservative and, therefore, acceptable.

4.8 Waste Gas System Failure

The changes made to this dose analysis, as documented in revisions to the BVPS-2 UFSAR, Section 15.7.1, and Tables 15.7-1 and 15.7-2, reflect corrected or conservative analysis input parameter values or input assumptions based on plant design and operation. The analysis methodology remained the same as had been previously reviewed and approved by the staff for BVPS-2.

The calculated offsite doses remain within 500 mrem whole body. The calculated control room doses remain within the dose guidelines of GDC-19. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the NRC staff finds the revisions to this DBA dose calculation, as documented in the licensee’s submittals, acceptable.

4.9 Summary

The NRC staff has reviewed the licensee’s revisions to the DBA dose analyses for BVPS-1 and 2. The revisions reflect corrected or conservative changes to input parameter values or assumptions based on plant design and operation. All calculated doses resulting from the revised dose analyses remain within applicable acceptance criteria. Additional changes were made to incorporate regulatory guidance such as GL 99-02. As a result of the conservative or corrective nature of the changes, and general compliance with regulatory guidelines such as the SRP and the DBA dose guidelines contained in 10 CFR Part 100 and 10 CFR Part 50,

Appendix A, GDC-19, the NRC staff finds the revisions to the DBA dose calculations for BVPS-1 and 2, as documented in the licensee's submittals, acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the *Federal Register* on March 15, 2001 (66 FR 15147). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 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.

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Attachments:
1. Table 1-1, "BVPS Unit 1 Design-Basis Accident Revised Offsite Doses (rem)," and Table 1-2, "BVPS Unit 1 Design-Basis Accident Revised Control Room Doses (rem)"
2. Table 2-1, "BVPS Unit 2 Design-Basis Accident Revised Offsite Doses (rem)," and Table 2-2, "BVPS Unit 2 Design-Basis Accident Revised Control Room Doses (rem)"

Date: March 22, 2001

Table 1-1
BVPS Unit 1 Design-Basis Accident Revised Offsite Doses (rem)

	Exclusion Area Boundary		Low Population Zone	
Postulated Accident	Thyroid	Whole Body	Thyroid	Whole Body
Fuel Handling Accident	25	0.58		
Waste Gas System				
Line Rupture	N/A	0.22	N/A	<0.2
Tank Rupture	N/A	<0.2	N/A	<0.2
SGTR				
Pre-accident Spike	1.3	<0.2		
Coincident Spike	1.4	<0.2		
Locked Rotor	5.8	0.47		
LOCA	200	4.3	14	0.4

Table 1-2
BVPS Unit 1 Design-Basis Accident Revised Control Room Doses (rem)

Postulated Accident	Thyroid	Whole Body
Fuel Handling Accident	6.3	<0.2
Waste Gas System		
Line Rupture	N/A	<0.2
Tank Rupture	N/A	<0.2
SGTR		
Pre-accident Spike	1.9	<0.2
Coincident Spike	3.1	<0.2
Rod Ejection	7.7	<0.2
Small Line Break	20	<0.2
LOCA	5.5	0.71

Table 2-1
BVPS Unit 2 Design-Basis Accident Revised Offsite Doses (rem)

	Exclusion Area Boundary		Low Population Zone	
Postulated Accident	Thyroid	Whole Body	Thyroid	Whole Body
Loss of AC	<1	<0.1	<1	<0.1
Locked Rotor	6.8	0.55	3.1	<0.1
Rod Ejection	24	<0.1	1.2	<0.1
Small Line Break	6.3	<0.1	<1	<0.1
SGTR				
Pre-accident Spike	17	<0.1	<1	<0.1
Coincident Spike	7.0	<0.1	<1	<0.1
LOCA	220	4.9	11	0.26
Waste Gas System				
Line Rupture	N/A	0.29		
Tank Rupture	N/A	<0.1		
MSLB				
Pre-accident Spike	16	<0.1	2.4	<0.1
Coincident Spike	27	<0.1	13	<0.1

Table 2-2
BVPS Unit 2 Design-Basis Accident Revised Control Room Doses (rem)

Postulated Accident	Thyroid	Whole Body
Loss of AC	<1	<0.2
Locked Rotor	7.5	<0.2
Rod Ejection	3.4	<0.2
Small Line Break	4.9	<0.2
SGTR		
Pre-accident Spike	2.3	<0.2
Coincident Spike	1.1	<0.2
LOCA	2	0.33
Waste Gas System		
Line Rupture	N/A	<0.2
Tank Rupture	N/A	<0.2
MSLB		
Pre-accident Spike	1.3	<0.2
Coincident Spike	2.8	<0.2