September 24, 2001

Mr. L. W. Myers
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 (BVPS-1 AND 2) - ISSUANCE OF AMENDMENT RE: 1.4-PERCENT POWER UPRATE AND REVISED BVPS-2 HEATUP AND COOLDOWN CURVES (TAC NOS. MB0996, MB0997, AND MB2557)

Dear Mr. Myers:

The Commission has issued the enclosed Amendment No. 243 to Facility Operating License No. DPR-66 and Amendment No. 122 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 January 18, 2001, as supplemented by letters dated February 20, April 12, May 7, May 18, June 9 (3 letters), June 26, June 29, August 21, and September 5, 2001.

These amendments approve revisions to the TSs and facility operating licenses to reflect increases in BVPS-1 and 2 maximum steady-state core power levels from 2652 megawatts thermal (MWt) to 2689 MWt, an increase of approximately 1.4 percent. The changes are anticipated to increase each unit's electrical output by approximately 12 MW. These increases are facilitated by the utilization of the Caldon Leading Edge Flowmeter for feedwater flow measurements. Revisions to BVPS-2 TS 3/4.4.9, "Pressure/Temperature Limits," heatup and cooldown curves are also approved. The Nuclear Regulatory Commission staff is deferring its review of the proposed changes associated with TS 3.7.1.1, "Main Steam Safety Valves (MSSVs)."

A copy of our safety evaluation is also enclosed. With respect to the portions of these amendments related to the 1.4-percent power uprate, a Notice of Issuance will be forwarded to

the Office of the Federal Register for publication. With respect to the portion of the amendment related to the revisions to BVPS-2 TS 3/4.4.9, “Pressure/Temperature Limits,” a 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. 243 to DPR-66
2. Amendment No. 122 to NPF-73
3. Safety Evaluation
4. Notice of Issuance

cc w/encls: See next page

the Office of the Federal Register for publication. With respect to the portion of the amendment related to the revisions to BVPS-2 TS 3/4.4.9, "Pressure/Temperature Limits," a Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Lawrence J. Burkhart, Project Manager, Section 1
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Division of Licensing Project Management
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cc w/encls: See next page

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ACCESSION NUMBER: **ML012490569** * SE input provided. No major changes.

OFFICE	PDI-1/PM	PDI-2/LA	SRXB/SC**	SPLB/SC*	SPSB/(A)SC*	EMCB/SC*	EMEB/SC*	EEIB/SC*
NAME	LBurkhart	MO'Brien	RCaruso for FAkstulewicz	GHubbard	PWilson	KWichman	KManoly	CHolden
DATE	9/19/01	9/19/01	9/10/01	4/24/01	3/28/01	7/8/01	9/6/01	4/2/01

OFFICE	EEIB/SC*	IOLB/SC*	TECH ED**	OGC**	PDI-1/(A)SC**	PDI/D	DLPM/D	ADPT/AD	NRR/D
NAME	EMarinos	DTrimble	PPuccio	MO'Neill	PTam	EAdensam	JZwolinski	BSheron	SCollins
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Beaver Valley Power Station, Units 1 and 2

Mary O'Reilly, Attorney
FirstEnergy Nuclear Operating Company
FirstEnergy Corporation
76 South Main Street
Akron, OH 44308

FirstEnergy Nuclear Operating Company
Regulatory Affairs Section
Thomas S. Cosgrove, Manager (2 Copies)
Beaver Valley Power Station
Post Office Box 4, BV-A
Shippingport, PA 15077

Commissioner James R. Lewis
West Virginia Division of Labor
749-B, Building No. 6
Capitol Complex
Charleston, WV 25305

Director, Utilities Department
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Director, Pennsylvania Emergency
Management Agency
2605 Interstate Dr.
Harrisburg, PA 17110-9364

Ohio EPA-DERR
ATTN: Zack A. Clayton
Post Office Box 1049
Columbus, OH 43266-0149

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

L. W. Pearce, Plant Manager (BV-IPAB)
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

Bureau of Radiation Protection
ATTN: Larry Ryan
P O Box 2063
Harrisburg, PA 17120

Mayor of the Borough of
Shippingport
P O Box 3
Shippingport, PA 15077

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 298
Shippingport, PA 15077

FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
ATTN: R. E. Donnellon, Director
Projects and Scheduling (BV-IPAB)
Post Office Box 4
Shippingport, PA 15077

Mr. J. A. Hultz, Manager
Projects & Support Services
FirstEnergy Corporation
76 South Main Street
Akron, OH 44308

FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mr. B. F. Sepelak
Post Office Box 4, BV-A
Shippingport, PA 15077

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. 243
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 January 18, 2001, as supplemented by letters dated February 20, April 12, May 7, May 18, June 9 (3 letters), June 26, June 29, August 21, and September 5, 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, the license is amended by changes to paragraph 2.C.(1) of Facility Operating License No. DPR-66, as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

   (1) Maximum Power Level

   FENOC is authorized to operate the facility at a steady state reactor core power level of 2689 megawatts thermal.

3. The license is also 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. 243, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. 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/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License No. DPR-66

Date of Issuance: September 24, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 243

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

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

Remove	Insert
Page 3	Page 3

Replace the following pages of 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
1-1	1-1
6-19	6-19

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. 122
License No. NPF-73

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

   C. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee) dated January 18, 2001, as supplemented by letters dated February 20, April 12, May 7, May 18, June 9 (3 letters), June 26, June 29, August 21, and September 5, 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, the license is amended by changes to paragraph 2.C.(1) of Facility Operating License No. NPF-73 as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

   (1) Maximum Power Level

   FENOC is authorized to operate the facility at a steady state reactor core power level of 2689 megawatts thermal.

3. The license is also 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. 122, 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.

4. 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/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License No. NPF-73

Date of Issuance: September 24, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 122

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

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

Remove	Insert
Page 3a	Page 3a

Replace the following pages of 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
1-1	1-1
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
3/4 4-32a	3/4 4-32a
3/4 4-32b	3/4 4-32b
3/4 4-32c	3/4 4-32c
3/4 4-32d	3/4 4-32d
6-19	6-19
6-20	6-20

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 243 AND 122 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 dated January 18, 2001 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML010230096), as supplemented by letters dated February 20 (ADAMS Accession No. ML010540305), April 12 (ADAMS Accession No. ML011130105), May 7 (ADAMS Accession No. ML011340076), May 18 (ADAMS Accession No. ML011440046), June 9 (3 letters) (ADAMS Accession Nos. ML011640192, ML011640189, and ML011640086), June 26 (ADAMS Accession No. ML011840215), June 29 (ADAMS Accession No. ML011870434), August 21, (ADAMS Accession No. ML012400228), and September 5, 2001 (ADAMS Accession No. ML012550393), the FirstEnergy Nuclear Operating Company (FENOC), et al., (the licensee) submitted a request for changes to the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2), facility operating licenses (FOLs) and Technical Specifications (TSs).

The requested changes included revisions to the BVPS-1 and 2 FOLs and TSs to reflect increases in the BVPS-1 and 2 maximum steady-state reactor core power levels from 2652 megawatts thermal (MWt) to 2689 MWt, an increase of approximately 1.4 percent. This power uprate is facilitated by using the Caldon, Inc. (Caldon), Leading Edge Flowmeter✔™ (LEFM✔™) and LEFM CheckPlus™ systems to measure feedwater flow at BVPS-1 and 2, respectively. The specific changes proposed in support of the power uprate include:

- Revision of Section 2.C.(1) of the BVPS-1 and 2 FOLs to reflect the increased maximum steady-state reactor core power level of 2689 MWt. In addition, the wording of this paragraph for BVPS-2 will be revised to be identical to that in the BVPS-1 FOL,

- Revision of TS 1.0, “Definitions,” to define RATED THERMAL POWER as a total reactor core heat transfer rate to the reactor coolant of 2689 MWt,”

- Revision to TS 6.9.6(b) to reflect the addition of 2 Caldon topical reports regarding the Caldon Leading Edge Flowmeter (LEFM) systems.

The licensee’s 1.4-percent power uprate request is based on a reduced uncertainty associated with measuring core thermal power using the newly installed Caldon LEFM systems to measure feedwater flow and temperature. The licensee has installed an LEFM✔™ system at BVPS-1 and its improved version, the LEFM CheckPlus™ system, at BVPS-2. The total power measurement uncertainty associated with utilization of the LEFM✔™ system is 0.6 percent, which can support a power uprate of up to 1.4 percent of rated thermal power. In its January 18, 2001, letter, the licensee referenced Caldon Topical Report ER-80P, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✔™ System,” and submitted Revision 2 of its supplement, Engineering Report ER-157P, “Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✔™ or CheckPlus™.” Caldon Topical Report ER-80P was approved by the Nuclear Regulatory Commission (NRC) staff on March 8, 1999 (Nuclear Documents System [NUDOCS] Accession No. 9903190065).

Tennessee Valley Authority, the licensee for Watts Bar Nuclear Plant, Unit 1, submitted Caldon Topical Report ER-160P, “Supplement to Topical Report ER-80P: Basis for Power Uprate with LEFM✔™ System,” in support of a power uprate request. In response to this amendment request, the measurement uncertainties associated with the LEFM✔™ system, as described in ER-160P, were approved by the NRC as documented in Amendment No. 31 to FOL No. NPF-90, dated January 19, 2001 (ADAMS Accession No. ML010260074), for Watts Bar Nuclear Plant, Unit 1. Additionally, the NRC staff approved the referencing of ER-160P in Amendment Nos. 194 and 169 to FOL Nos. NPF-14 and NPF-22, dated July 6, 2001 (ADAMS Accession No. ML011760551), for Susquehanna Steam Electric Station, Units 1 and 2, respectively.

The licensee’s January 18, 2001, amendment request credited a lower uncertainty as described in Caldon Topical Report ER-157P for the LEFM CheckPlus™ system than that previously approved for the LEFM✔™ system. The alleged increase in accuracy of the LEFM CheckPlus™ system could further reduce the total power measurement uncertainty and, if supported by adequate plant-specific information, could support a higher than 1.4-percent power uprate. Due to the lack of an NRC staff review of the uncertainties associated with the LEFM CheckPlus™ system, the licensee’s request for only a 1.4-percent power uprate, and the previously approved 1.4-percent power uprate amendments and associated Caldon Topical Reports ER-80P and ER-160P, the licensee decided to provide sufficient information to justify crediting the same uncertainty for the LEFM CheckPlus™ system as that previously approved for the LEFM✔™ system. The licensee provided this information and the revised plant-specific uncertainties by letters dated April 12 and June 9, 2001. In addition, in its letter dated September 5, 2001, the licensee replaced the reference to Caldon Topical Report ER-157P with a reference to ER-160P. Consequently, the NRC staff did not review Topical Report ER-157P.

The plant-specific instrument uncertainties supporting the proposed BVPS-1 and 2 power uprates were reviewed under separate amendment requests dated December 27, 2000

(ADAMS Accession No. ML003782095). These requests were approved by the NRC staff on July 20, 2001, by Amendment Nos. 239 and 120 to FOL Nos. DPR-66 and NPF-73 for BVPS-1 and 2, respectively (ADAMS Accession No. ML011910223). These amendments approved, among other things, the implementation of the revised thermal design procedure (RTDP) and revisions to reactor trip system and engineered safety feature actuation system (ESFAS) trip setpoints and allowable values. In support of requests associated with Amendment Nos. 239 and 120, the plant-specific instrument uncertainty information was provided in Westinghouse Topical Reports "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 1, WCAP-15264, Revision 3, December 2000" (WCAP-15264), and "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 2, WCAP-15265, Revision 3, December 2000" (WCAP-15265). These topical reports were submitted via letter dated December 27, 2000 (ADAMS Accession No. ML003782095), as supplemented by letter dated June 9, 2001 (ADAMS Accession No. ML011640086).

The licensee also requested changes to BVPS-2 TS 3/4.4.9, "Pressure/Temperature Limits," heatup and cooldown curves (Figures 3.4-2 and 3.4-3) to reflect a reduction in the applicability of the current limits from 15 effective full-power years (EFPYs) to 14 EFPYs. This reduction is consistent with the increased neutron fluence associated with the proposed increased power level.

Additionally, in its January 18, 2001, letter, the licensee requested changes to BVPS-1 and 2 TS 3.7.1.1, "Main Steam Safety Valves (MSSVs)," to reflect consistency with technical specification traveler form 235 (TSTF-235), Revision 1, and the improved standard TSs. The NRC staff is deferring its review of this change and will forward its related evaluation by separate correspondence.

With respect to the power uprate portions of these amendment requests, the February 2, April 12, May 7, May 18, June 9 (3 letters), June 26, June 29, August 21, and September 5, 2001, letters provided clarifying information that did not change the scope of the original *Federal Register* notice.

With respect to the changes to BVPS-2 TS 3/4.4.9, the February 2, April 12, May 7, May 18, June 9 (3 letters), June 26, June 29, August 21, and September 5, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified core thermal power, and the uncertainty of the calculated values of this thermal power determines the probability of exceeding the power levels assumed in the design-basis transient and accident analyses. In this regard, Appendix K to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 requires loss-of-coolant accident (LOCA) and emergency core cooling system (ECCS) analyses to assume that the reactor has been operating continuously at a power level at least 102 percent of the licensed thermal power to allow for uncertainties, such as instrument error. The phrase "such as" suggests that the 2 percent power margin was intended to address

uncertainties related to heat sources in addition to the instrument measurement uncertainties. As documented in the *Federal Register* on June 1, 2000 (65 FR 34916), the NRC concluded that, at the time of the original ECCS rulemaking, the 2-percent power margin requirement was solely based on the considerations associated with reactor power measurement uncertainty. This development could justify a reduced margin between the licensed power level and the power level assumed in the ECCS analysis and, therefore, a power uprate.

In order to reduce an unnecessarily burdensome regulatory requirement and to avoid unnecessary exemption requests, the Commission published the final rule in the *Federal Register* on June 1, 2000. This final rule allows licensees the option of justifying a smaller margin of power measurement uncertainty by using more accurate instrumentation to calculate the reactor thermal power or maintaining the current margin of 2-percent power. Licensees may apply the reduced margin to operate the plant at a level higher than the current licensed power or use the margin to relax ECCS-related TSs. The final rule, by itself, does not allow licensees to increase the licensed power level without NRC staff approval. Since the licensed power level of a nuclear power plant is a specified limit in the TSs and often in the FOL, the proposals to increase the licensed power level must be reviewed and approved through the license amendment process. Such license amendment requests should include a justification for the reduced power measurement uncertainty to support the proposed power uprate.

3.0 EVALUATION

3.1 LEFM - Instrumentation and Control

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system (NSSS). This calculation is called the “secondary calorimetric” for a pressurized water reactor (PWR). The accuracy of this calculation primarily depends on the accuracy of feedwater flow and feedwater net enthalpy measurements. Thus, an accurate measurement of feedwater flow and temperature will result in an accurate calorimetric calculation and an accurate calibration of the nuclear instrumentation.

The instrumentation for measuring feedwater flow typically uses an orifice plate, a venturi meter, or a flow nozzle to generate a differential pressure proportional to the feedwater velocity in the pipe. Of the three differential pressure devices, a venturi meter is most widely used for feedwater flow measurement in nuclear power plants. The feedwater temperature is typically measured by resistance temperature detectors (RTDs). The BVPS-1 and 2 designs use a venturi meters for flow measurement and RTDs for temperature measurement in each steam generator feedwater system.

The major advantage of a venturi meter is a relatively low head loss as the fluid passes through the device. The major disadvantage of the device is fouling, which causes the meter to indicate a higher differential pressure and hence a higher than actual flow rate. This leads the plant operator to calibrate nuclear instrumentation high, which is conservative with respect to reactor safety, but causes the electrical output to be proportionally low when the plant is operated at its thermal power rating. To eliminate the fouling effects, the flow device is removed, cleaned, and re-calibrated. Due to the high cost of re-calibration and the need to improve flow instrumentation uncertainty, the industry assessed other flow measurement techniques and found the LEFM systems to be a viable alternative.

The Caldon Chordal LEFM, which is employed in the LEFM✔™ and CheckPlus™ systems, is an ultrasonic flowmeter that uses acoustic energy pulses to determine the feedwater mass flow rate and temperature. The meter is based on transit time technology. The transit time technology sends an ultrasonic signal diagonally through the fluid and then measures the time it takes to travel upstream and downstream. The sound travels faster when the pulse traverses the pipe with the flow and slower when the pulse traverses the pipe against the flow. The difference in these times is proportional to the velocity of the fluid in the pipe. The LEFM uses these transit times and the time differences between pulses to determine the fluid velocity and temperature (temperature of pure water can be determined from its sound velocity and pressure).

At BVPS-1 and 2, the LEFM systems consists of an electronic cabinet in the process control area of the plant and a measurement section (spool piece) that is permanently installed in the 26-inch main feedwater header. It is a single digital system controlled by software using the ultrasonic transit time method to measure four line integral velocities at precise locations with respect to the pipe center line. The system numerically integrates the four measured velocities to determine the mass flow rate and fluid temperature. These measurements are used by the plant computer to determine the reactor thermal output.

The LEFM✔™ system uses eight transducers in a configuration of two on each of the four acoustic measurement paths in a single plane of the spool piece. The LEFM CheckPlus™ system uses sixteen transducers in a similar configuration in two orthogonal planes of the spool piece. As such, the LEFM CheckPlus™ system is a combination of two LEFM✔™ systems taking the average of two numerical integrations of four measurements each in two orthogonal planes. This measurement is inherently more accurate than the integration of four measurements in a single plane by the LEFM✔™ system and, therefore, should provide a better measurement accuracy. Also, due to twice as many measurements by the LEFM CheckPlus™ system, the error in the statistical treatment of the instrumentation uncertainty is reduced. However, the licensee did not take credit for the improved accuracy and stated only that the LEFM CheckPlus™ system provides feedwater flow measurement that is at least as accurate as that provided by the NRC-approved LEFM✔™ system. In its safety evaluation supporting Amendment Nos. 239 and 120 to FOL Nos. DPR-66 and NPF-73, dated July 20, 2001, the NRC staff found that the licensee provided sufficient justification to allow the same feedwater flow measurement uncertainty value for the LEFM CheckPlus™ system as for the LEFM✔™ system.

The licensee stated that the system's software was developed and will be maintained under a verification and validation (V&V) program that complies with the Institute of Electrical and Electronics Engineers (IEEE) Standard 7-4.2-1990 and American Society of Mechanical Engineers (ASME) Standard NQA-2A-1990. The V&V program has been applied to all system software and hardware, and includes a detailed code review. The LEFM system's indications of feedwater flow and temperature will be displayed on the local display panel and transmitted to the plant process computer. This information will be directly substituted for the venturi-based flow indications and the RTD temperature indications that are currently used in the plant calorimetric calculation. However, the licensee will continue to use the venturi-based feedwater flow measurement for feedwater control and other functions. A real-time display of thermal power using the LEFM system will be available in the main control room, and an audible alarm will annunciate to the operator when the LEFM system is not operating within its design-basis accuracy. The licensee stated that the LEFM system provides an on-line verification of the

accuracy of the feedwater flow and temperature measurements and will significantly improve measurement accuracy and reliability.

Caldon Topical Report ER-80P and its supplement, Topical Report ER-160P (previously approved by the NRC staff) describe the LEFM✔™ system and include calculation of thermal power uncertainties for a typical 2-loop PWR or BWR using the LEFM✔™ system for feedwater flow and temperature measurement. The calculation results for a typical PWR or BWR show a total thermal power determination uncertainty of ±0.6 percent with a 95-percent confidence limit. The report provides a generic basis for the proposed 1.4-percent uprate of the licensed reactor power and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties.

The NRC staff's safety evaluation (SE), dated March 8, 1999, regarding the Caldon Topical Report ER-80P, included four additional requirements to be addressed by a licensee requesting a power uprate. FENOC's submittals addressed each of the four requirements as follows:

1. The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

This item was addressed in the licensee's letters dated January 18, February 20, and May 18, 2001. FENOC stated that implementation of the power uprate will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level and will be based on the Caldon's recommendations with the requirement that Caldon notify FENOC of any deficiency that could affect the design-basis accuracy of the LEFM systems. BVPS-1 and 2 have procedures, as required by 10 CFR, Part 50, Appendix B, for receiving and addressing vendor's deficiency reports, reporting deficiencies to the vendor, and performing the necessary and recommended corrective actions. The LEFM system software will be maintained under Caldon's V&V program, and all other instrumentation affecting the power calorimetric, including the plant process computer, will continue to be maintained in accordance with the existing plant maintenance and calibration procedures. The LEFM system operability requirements will be included in the BVPS-1 and 2 licensing requirements manual (LRM). If the LEFM system is not operable, the LRM will require the reduction of power to less than or equal to the pre-uprate maximum power level (2652 MWt) and the use of the venturi measurement until the LEFM system is returned to operable status.

The licensee will use these procedures to control configuration and calibration of both 1E and non-1E instruments. The calibration frequency of the instrumentation affecting the power calorimetric is based on an 18-month refueling outage with the exception that the BVPS-2 steam generator blowdown flow transmitter, which will be calibrated once every 6 months with a ± 25-percent grace period. This 6-month calibration frequency shall remain in effect until such time that an 18-month calibration frequency can be justified by either accumulation of sufficient data or installation of more accurate instrumentation.

2. For plants that currently have an LEFM system installed, the licensee should provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

BVPS-1 and 2 have never used an LEFM system; therefore, neither unit has any operational or maintenance history to evaluate.

3. The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternate methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installation for comparison.

FENOC referenced Revision 3 of Westinghouse Topical Reports, WCAP-15264, and WCAP-15265. These two topical reports document the Westinghouse Revised Thermal Design Procedures (RTDP) and include the plant-specific power calorimetric measurement uncertainty calculations for BVPS-1 and 2, respectively. The calculation methodology complies with the recommendations of ANSI/ISA Standard 67.04 and Regulatory Guide 1.105, Revision 2. This methodology has been reviewed and approved by the NRC staff for Westinghouse PWRs. In these calculations, Westinghouse statistically combined LEFM system uncertainty with other instrumentation uncertainties affecting the plant power calorimetric uncertainty. The resulting 3-loop power calorimetric measurement uncertainty for BVPS-1 and 2 was found to be 0.6 percent of the rated thermal power, which justifies the proposed 1.4-percent power uprate. The NRC staff finds that the licensee's calculation is based on an accepted plant setpoint methodology and, therefore, is acceptable.

4. Licensees of plants where the ultrasonic meter (including the LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), should provide additional justification for use. This justification should show either that the meter installation is independent of the plant-specific flow profile for the stated accuracy or that the installation can be shown to be equivalent to known calibrations and the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM system installation and calibration assumptions.

In its letter dated February 20, 2001, FENOC stated that the BVPS-1 and 2 flow elements were calibrated to the site-specific piping configuration at Alden Research Laboratory (ARL). The ARL tests were performed on the LEFM system spool pieces installed in a laboratory model hydraulic geometry of each BVPS unit. The licensee stated that the final acceptance of the site-specific uncertainty analysis will occur after completion of the commissioning process to confirm that the actual performance in the field meets the uncertainty bounds established in the Westinghouse topical reports.

The NRC staff finds that FENOC's response has sufficiently resolved the plant-specific concerns regarding maintenance and calibration of the LEFM systems and other instrumentation affecting the power calorimetric, hydraulic configuration of the installed LEFM systems, processes and contingencies for an inoperable LEFM system, and methodology for calculating the LEFM system measurement uncertainty and the plant power calorimetric uncertainty.

Based on the review of the licensee's submittals on the LEFM✔™ and LEFM CheckPlus™ systems and plant power calorimetric uncertainty, the NRC staff finds that the BVPS-1 and 2 thermal power measurement uncertainty, with the LEFM✔™ system in BVPS-1 and the LEFM CheckPlus™ system in BVPS-2, is limited to ±0.6 percent of the reactor thermal power and can support the proposed 1.4-percent thermal power uprate of each unit. The NRC staff also finds that the licensee adequately addressed the four additional requirements outlined in the NRC staff SE on the LEFM✔™ system Topical Report.

3.2 Nuclear Steam Supply System (NSSS) Design Parameters

The NSSS design parameters provide the reactor coolant system (RCS) and secondary system conditions for use in the NSSS analyses and evaluations. FENOC presented the new parameters for the power uprate and incorporated them into its evaluations. The modified input assumptions include an increased NSSS power level of 2697 MWt and a corresponding increase in the feedwater temperature to 439.3 °F. The NSSS power is the summation of the reactor core power of 2689 MWt and the thermal power generated by the reactor coolant pumps (RCPs) of 8 MWt. Other parameters that change as a result of the uprate include the RCS temperatures: RCS hot leg temperature (T_{hot}) increased by 0.4 °F and RCS cold leg temperature (T_{cold}) decreased by 0.4 °F. And finally, there were small changes in the following secondary-side parameters: steam temperature (T_{steam}), steam pressure (P_{steam}), and steam mass flow rate (M_{steam}). The NRC staff evaluated these changes to the plant conditions and finds them to adequately represent the uprated plant behavior; therefore, the NSSS design parameters are acceptable.

3.3 Design Transients

3.3.1 NSSS Design Transients

To support the power increase for BVPS-1 and 2, FENOC reviewed the current primary- and secondary-side design transients to determine their continued applicability at the uprated design conditions. The licensee compared the current analysis parameters to those expected at the uprated conditions. When the existing parameters remained bounding, the licensee concluded that the design transient analyses remain valid.

The NRC staff has reviewed the licensee's methodology for determining and re-evaluating the initial design transient parameters. The NRC staff finds that the methodology is conservative and, therefore, is acceptable.

3.3.2 Auxiliary Equipment Design Transients

The licensee reviewed the NSSS auxiliary equipment design transients by comparing the revised operating conditions to the current transient conditions. FENOC determined that the only transients affected by the uprate were the temperature transients, i.e., those impacted by the changes to the full-load NSSS operating temperatures, T_{hot} and T_{cold}. The temperature transients for BVPS-1 and 2 were previously analyzed for a worst-case T_{hot} of 630 °F and T_{cold} of 560 °F. The uprated T_{hot} and T_{cold} are 610.8 °F and 541.6 °F, respectively. These temperatures remain within the limits set in the licensee's transient analyses; therefore, the NRC staff finds that the current auxiliary equipment design transient analyses remain acceptable for the power uprate.

3.4 NSSSs

3.4.1 RCS

The licensee performed various assessments and demonstrated that the RCS design-basis functions would be met at the revised design conditions. Based on these assessments, the NRC staff provides the following conclusions.

The major components of the main steam system support the increased heat transfer requirements. The residual heat removal system adequately removes the increased decay heat. The RCS control and protection functions are not significantly affected. The RCS thermal design flow does not change as a result of the uprate. The pressurizer spray flow rate of 600 gallons-per-minute (gpm) is still achievable. The RCS design temperature and pressure of 650 ^{o}F and 2485 pounds-per-square inch gauge (psig) continue to remain bounding. The pressurizer design temperature and pressure of 680 ^{o}F and 2485 psig continue to remain applicable. Finally, the pressurizer relief tank sizing and setpoint, pressurizer relief valve sizing and discharge piping pressure drop, pressurizer relief valve inlet pressure drop, and pressurizer surge line pressure drop parameters are not affected by the power uprate.

On the basis of these conclusions, the NRC staff finds that the RCS remains acceptable for the power uprate.

3.4.2 Safety Injection System (SIS)

The revised design conditions have no direct effect on the overall performance capability of the SIS. This system will continue to deliver the safety injection flow at the design-basis RCS and containment pressures since there are no changes in the RCS operating pressure. Therefore, the NRC staff finds the SIS acceptable for the power uprate.

3.4.3 Chemical and Volume Control System (CVCS)

The primary role of the CVCS is to manage RCS water inventory, boron concentration and water chemistry. In order to meet these requirements, the CVCS has to perform the following functions: (1) add boric acid and corrosion control chemicals to the RCS, (2) cleanup, degasify and add makeup to the coolant, and (3) reprocess the letdown water from the RCS and the RCP seal water injection. The power uprate will change the operating temperatures of the RCS, which could have some impact on these CVCS functions. However, the licensee's analysis indicates that after power uprate, the maximum cold leg temperature will be 541.6 ^{o}F, which is lower than the design system inlet temperature and much lower than the shell-side design temperature for the regenerative heat exchanger. Also, the excess letdown heat exchanger inlet temperature is less than the design inlet operating temperatures of 543.5 ^{o}F and 547 ^{o}F for BVPS-1 and 2, respectively. The result is a lower excess letdown outlet temperature. Since the excess letdown path is primarily used for RCP seal injection, when normal letdown flow is not available, a lower seal water temperature after power uprate will not cause any RCP problems.

The CVCS equipment will not require modification to account for the power uprate, and the reevaluations of water volumes and boric acid concentration will be performed as part of a

normal reload safety evaluation process. A very slight increase of N-16 activity will have only a negligible effect on the radioactivity of fluids in the excess lines. As a result, the licensee concluded that the proposed power uprate will not have any deleterious effects on the operation of the CVCS. The NRC staff reviewed the licensee's analysis and concurs with the licensee's conclusion. The NRC staff finds the licensee's analysis appropriate and conservative and finds the CVCS system acceptable for the power uprate.

3.4.4 Residual Heat Removal (RHR) System

The RHR system is designed to remove sensible and decay heat from the core and to reduce the temperature of the RCS during the second phase of plant cooldown. As a secondary function, the RHR system is used to transfer refueling water between the refueling water storage tank and the refueling cavity at the beginning and end of refueling operations.

The licensee performed an evaluation of the effects of the power uprate on the design-basis operation of the RHR system and concluded that the existing design-basis analysis is acceptable for the power uprate.

Based on the information provided by the licensee, the NRC staff finds that the licensee's evaluation is conservative and acceptable. Therefore, the NRC staff finds the RHR system acceptable for the power uprate.

3.4.5 Spent Fuel Pool Cooling System

The BVPS-1 and 2 spent fuel pool cooling system removes the decay heat generated by the stored fuel assemblies. The licensee analyzed the thermal-hydraulic effects on the spent fuel pool cooling system due to the power uprate and determined that the proposed power uprate will have little or no impact on the performance of this system.

Based on the information provided by the licensee and the experience gained from prior NRC staff reviews of other power uprate applications for similar PWRs, the NRC staff concurs with the licensee that operations at the proposed 1.4-percent uprate power level will have little impact on the operation of this system. Therefore, the NRC staff finds the spent fuel pool cooling system acceptable for the power uprate.

3.4.6 NSSS Control Systems

The licensee evaluated the following transients to ensure that the plants would respond without generating a reactor trip system or an ESFAS actuation. These transients include the 10-percent step load increase, 10-percent step load decrease, 50-percent load rejection, and 5-percent-per-minute ramp load increase. The current analyses for these transients include a 2-percent power calorimetric uncertainty. Since the Caldon LEFM systems reduce the calorimetric uncertainty to no more than 0.6 percent, the initial uncertainty assumption bounds the 1.4-percent uprate, and the initial analyses remain valid. Therefore, the NRC staff finds the analyses acceptable for the power uprate conditions.

Similarly, the pressurizer power-operated relief valve and spray valve capacities for response to operational transients were also analyzed with a 2-percent power uncertainty. These analyses demonstrated that the valve capacities were adequate for the transients. Since the valves were

adequate in the original analyses, and since the 2-percent power uncertainty in the analyses bounds the 1.4-percent power uprate, the NRC staff finds that the pressurizer power operated relief valves and spray valve capacities are acceptable for the power uprate.

The licensee also examined the rod and steam dump control system stability for key operational transients. These examinations determined that the stability of these control system is not a function of the power level or full-load average coolant temperature (T_{avg}). Rather it is a function of the rod and steam dump control system setpoints and the reactor core kinetics. FENOC concluded that since the 1.4-percent uprate did not change the control system setpoints and did not significantly change the core kinetics, the rod and steam dump control system stability is not affected by the uprate. The NRC staff agrees with the licensee's conclusion that the power uprate does not result in significant changes to core kinetics and that no modifications to the rod and steam dump control system setpoints are required. Therefore, the NRC staff finds the NSSS control systems acceptable for the power uprate.

3.4.7 Cold Overpressure Mitigation System (COMS)

The COMS is designed to protect the RCS from overpressure events when the RCS temperature is below 329 ºF and 350 ºF for BVPS-1 and 2, respectively. Changes to full-power operating parameters, such as NSSS power, do not impact the COMS. Thus, the existing COMS analysis remains bounding. Since changes to the full-power operating parameters do not impact the COMS, the NRC staff finds the COMS acceptable for the power uprate.

3.5 NSSS Components

3.5.1 Reactor Vessel Structural Evaluation

The licensee reported that the power increase will result in the revised design parameters given in Tables 3-1 of Enclosure 1 to FENOC's letter dated January 18, 2001, and discussed in Section 3.2 of this SE. However, the RCS pressure remains unchanged. The licensee indicated that the design-basis analyses of record for the existing LOCA, jet impingement, and thrust loads remain bounding for the proposed power uprate conditions.

The licensee evaluated the reactor vessel for the effects of the revised design-basis input parameters on the most limiting vessel locations with regard to ranges of stress intensity and fatigue usage factors in each of the regions, as identified in the reactor vessel stress reports. These parameters include the limiting T_{hot} and T_{cold} and the NSSS design transients. As a result of the proposed power uprate, T_{hot} will increase by 0.4 °F and T_{cold} will decrease by 0.4 °F. The licensee indicated that the small changes in T_{hot} and T_{cold} are within the existing margins in the current BVPS-1 and 2 reactor vessel design-basis analyses. Also, the current design-basis transients will remain unchanged for the uprated conditions. Therefore, the existing reactor vessel structural analyses remain bounding for the power uprate conditions. The licensee also concluded that the stress intensities and cumulative usage factors (CUFs) of the reactor vessel components will continue to satisfy the limits specified in the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, 1968 Edition, with addenda through winter 1968 for BVPS-1, and the 1971 Edition with addenda through summer 1972 for BVPS-2, which are the codes of record.

The NRC staff finds that the licensee's evaluation is bounded by the licensing code of record and the original design basis; and, therefore, concludes that the reactor vessel is acceptable for the power uprate.

3.5.2 Reactor Vessel Integrity

3.5.2.1 Design Conditions and Neutron Fluence Changes

The licensee evaluated the reactor vessel integrity analysis for the 1.4-percent power uprate by examining the revised design conditions and the increase in neutron fluences. With respect to the revised design conditions, T_{cold} increases slightly to 541.6 °F due to the 1.4-percent power uprate. The current reactor vessel integrity analysis assumes that T_{cold} is maintained between 530 °F and 590 °F. Therefore, the temperature assumption for the analysis is not affected.

With respect to the effects of the neutron fluence, the licensee evaluated the projected neutron fluence on the vessel for the uprated power level. These fluence projections serve as input to the reactor vessel integrity evaluations. Specifically, fluence values are used to evaluate the end-of-life (EOL) transition temperature shift for development of the surveillance capsule withdrawal schedules, determine EOL upper shelf energy (USE) values, adjust reference temperature values for determining the applicability of the heatup and cooldown curves, adjust Emergency Response Guideline (ERG) limits, and determine pressurized thermal shock (RT_{PTS}) values. An evaluation of the neutron exposure of the reactor vessel materials was also performed to bound the effects of the proposed 1.4-percent increase in power. This evaluation included assessing the locations of maximum exposure at the inner diameter of the vessel, as well as axial, azimuthal, and radial locations throughout the vessel wall.

The fast neutron exposure levels are defined at depths within the vessel wall equal to 25 and 75 percent of the wall thickness for each of the materials constituting the beltline region. This is done to satisfy the requirements of 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," for the calculation of pressure/temperature limit curves for normal heatup and cooldown of the RCS. These locations are commonly referred to as the 1/4T and 3/4T positions in the vessel wall. The 1/4T exposure levels are also used in the determination of upper shelf fracture toughness as specified in 10 CFR Part 50, Appendix G. Maximum neutron exposure levels experienced by each of the beltline materials are required to determine the RT_{PTS} values, which are then compared with the applicable pressurized thermal shock screening criterion as defined in 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." The maximum exposure levels occur at the inner radius of the vessel.

The results of the fast neutron exposure evaluations for BVPS-1 and 2 account for the uprated power level. The results are derived using the conservative assumption that the power uprate is initiated coincident with the last surveillance capsule withdrawal (capsule Y for BVPS-1 and capsule V for BVPS-2) from each unit. The resulting fast neutron (E > 1.0 MeV) exposure projections increase due to the power uprate. The new projections were used by the licensee as inputs to the reactor vessel integrity evaluations.

The NRC staff finds that the licensee's analysis conservatively accounts for the revised design conditions and neutron fluence and, therefore, is acceptable. Therefore, the NRC staff finds the licensee's evaluation, with respect to reactor vessel integrity, acceptable for the power uprate.

3.5.2.2 Surveillance Capsule Withdrawal Schedule

The licensee has developed a schedule to periodically withdraw the surveillance capsules from the reactor vessel to effectively monitor the condition of the reactor vessel materials under actual operating conditions. This surveillance capsule withdrawal schedule is consistent with American Society for Testing and Materials (ASTM) E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." The licensee developed a surveillance capsule withdrawal schedule for the BVPS-1 reactor vessels based on the projected neutron fluence values resulting from the 1.4-percent power uprate. The withdrawal of a capsule was scheduled at the nearest vessel refueling outage to the calculated EFPYs. The licensee concluded that no change is needed to the current withdrawal schedules for BVPS-1 and 2 as a result of the projected neutron fluence values. The NRC staff finds this conservative and, therefore, acceptable.

3.5.2.3 Heatup and Cooldown Pressure/Temperature (PT) Limit Curves

The licensee evaluated the BVPS-1 power uprate condition for the 16 EFPY heatup and cooldown curves and determined that the curves are not affected by the 1.4-percent power uprate. Capsule Y was withdrawn in BVPS-1 refueling outage 13 in the spring of 2000. A revised evaluation must be submitted within 1 year of capsule withdrawal per 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." Based on the information provided by the licensee, the NRC staff finds that the licensee conservatively evaluated the effects of the power uprate on the applicability of the heatup and cooldown PT limit curves for BVPS-1. Therefore, the NRC staff finds these curves acceptable for the power uprate.

For BVPS-2 an evaluation of the current 15 EFPY heatup and cooldown curves was performed to determine if a change in EFPY was required due to the uprated fluence values. The heatup and cooldown curves are documented in WCAP-15139, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves During Normal Operation at 15 EFPY Using Code Case N-626." Code Case N-626 has since been re-designated as Code Case N-640. The heatup and cooldown curves documented in WCAP-15139 were generated using the most limiting adjusted reference temperature (ART) values and the NRC-approved methodology documented in WCAP-14040-NP-A, Revision 2, with two exceptions. These exceptions are:

- The fluence values are calculated fluence values, not the best-estimate fluence values.
- The K_{IC} critical stress intensities are used in place of the K_{Ia} critical stress intensities, based on the approved methodology in ASME Code Case N-640.

Given the results of its evaluation, the licensee concluded that the current heatup and cooldown curves remain applicable to 14 EFPYs with the 1.4-percent power uprate. Consequently, the licensee requested that the NRC staff approve the proposed revision to the TS 3/4.4.9, "Pressure/Temperature Limits," heatup and cooldown curves applicability from 15 EFPYs to

14 EFPYs to accommodate the uprated conditions. In determining the revised applicability of the curves, the licensee conservatively assumed that the increased fluence commenced after removal of surveillance capsule V, which is earlier than the actual date of the implementation of the power uprate. The NRC staff finds that the changes in the applicability of the heatup and cooldown curves from 15 EFPYs to 14 EFPYs in BVPS-2 TS 3/4.4.9 conservatively account for the effects of the power uprate and are, therefore, acceptable.

3.5.2.4 Pressurized Thermal Shock (PTS)

The licensee established RT_{PTS} screening criteria values (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates, and beltline circumferential weld seams for end-of-license operation based on the NRC screening criteria for PTS (10 CFR 50.61). The RT_{PTS} values for beltline region materials of the BVPS-1 and 2 reactor vessels for end of license (28 EFPY for BVPS-1 and 32 EFPY for BVPS-2) were recalculated for the 1.4-percent uprate. These RT_{PTS} values increase due to the 1.4-percent power uprate. However, the BVPS-1 and 2 RT_{PTS} values remain below the NRC screening criteria values through 28 EFPY for BVPS-1 and 32 EFPY for BVPS-2. The NRC staff reviewed the licensee evaluation and concurs with its conclusion, and finds the effects on PTS acceptable for the power uprate.

3.5.2.5 Emergency Response Guideline (ERG) Limits

The licensee determined the new RT_{PTS} values for BVPS-1 and 2 based on the bounding fluence projections. A comparison of the current RT_{PTS} calculation (which is the RT_{NDT} value at the end-of-license [28 EFPY for BVPS-11 and 32 EFPY for BVPS-2]) to the uprated RT_{PTS} values for BVPS-1 and 2 was made to determine if the applicable ERG category (Westinghouse Owners Group Emergency Response Guidelines, Rev. 1C, September 30, 1997) would change.

The most limiting RT_{PTS}, value for BVPS-1 is 259 °F at 28 EFPY. The BVPS-1 limiting material is the lower shell plate B6903-1. The ERGs were developed for three specific categories. The first two categories were developed for an axial flaw in a longitudinal weld, plate or forging. The third category was developed for a circumferential weld flaw with an ART greater than 250 °F. BVPS-1 steam generators would be in Category II through approximately 21.7 EFPY. The ERG limit for operation beyond 21.7 EFPY will need to be based on a plant-specific evaluation.

The most limiting RT_{PTS} value for BVPS-2 is 15 °F at 32 EFPY. Since this value is well below the 200 °F maximum for Category I ERG limits, the BVPS-2 ERG plant-specific limits for current EOL (32 EFPY) remain valid for the 1.4-percent uprate. Therefore, the NRC staff finds the licensee's evaluation acceptable for the power uprate.

3.5.2.6 Upper Shelf Energy (USE)

Since the bounding neutron fluence values for the 1.4-percent uprate have increased, the USE values were recalculated for BVPS-1 and 2. The licensee determined that the reactor beltline materials in both reactor vessels are expected to have a USE greater than 50 foot-pounds (ft-lb) through the end-of-license (28 EFPY for BVPS-1 and 32 EFPY for BVPS-2) as required by 10 CFR Part 50, Appendix G. The NRC staff finds that the licensee's analysis is

conservative and the results meet the guidelines of 10 CFR Part 50, Appendix G. Therefore, the NRC staff finds the revised USE values acceptable for the power uprate.

3.5.2.7 Reactor Vessel Integrity Summary

The NRC staff finds that the licensee has conservatively accounted for the effects of the power uprate regarding these reactor vessel integrity issues. The NRC staff finds that the licensee's evaluations associated with the effects of the power uprate on the surveillance withdrawal schedules, heatup and cooldown PT limit curves (including the TS changes to the applicability BVPS-2 curves), PTS, ERG limits, and USE values, are acceptable. Therefore, the NRC staff finds the reactor vessel integrity acceptable for the power uprate.

3.5.3 Reactor Core Support Structures and Vessel Internals

By letters dated January 18 and August 21, 2001, the licensee provided information regarding its evaluation of the reactor vessel core support and internal structures. The limiting reactor internal components evaluated include the lower core plate, core barrel, baffle plate, baffle/barrel region bolts, and upper core plate. The licensee indicated that because the reactor internal components were not licensed to the ASME B&PV Code, the evaluation was conducted in compliance with the original Westinghouse design criteria as documented in the BVPS-1 and 2 updated final safety analysis reports (UFSARs). In addition, the licensee indicated that the ASME Code 1989 Edition was used for the evaluation of the lower core plate for the proposed 1.4-percent power uprate.

The licensee evaluated these critical reactor internal components considering the revised design conditions provided in Table 3-1 of Enclosure 1 to the January 18, 2001, letter. The licensee indicated that the 1.4-percent uprate does not change the current design-basis seismic and LOCA loads. For the baffle/barrel region and the upper core plate, the current structural and thermal analyses of record for BVPS-1 and 2 remain bounding for the power uprate condition. The licensee also indicated that maximum stresses and CUFs for the reactor internal components remain less than the acceptable limits. The other reactor internal components are less limiting. In addition, the licensee indicated that the current analysis of record for flow-induced vibration remains bounding for the power uprate. As a result of these evaluations, the licensee concluded that the reactor internal components at BVPS-1 and 2 will be structurally adequate for the proposed power uprate conditions.

The NRC staff finds that the licensee's evaluation is bounded by the licensing code of record and the original design basis; and, therefore, the reactor core support structure and vessel internals are acceptable for the power uprate.

3.5.4 Control Rod Drive Mechanisms (CRDMs)

The Model L-106A CRDMs and the capped latch housings are installed in the BVPS-1 and 2 reactor vessel heads. The licensee evaluated the adequacy of the CRDMs by reviewing the BVPS-1 and 2 CRDM design specifications and stress reports to compare the design-basis input parameters against the revised design conditions in Table 3-1 of Enclosure 1 to FENOC's January 18, 2001, letter. The T_{hot} is considered the most limiting for the CRDM design. The comparison indicated that the T_{hot} for the 1.4-percent power uprate is bounded by the design-basis analysis. The licensee also indicated that the NSSS design transients used in the original

design-basis analysis remain bounding for the 1.4-percent power uprate. Therefore, the licensee concluded that the existing design-basis analysis for BVPS-1 and 2 CRDM components' stresses and CUFs remain valid for the proposed 1.4-percent power uprate conditions.

The NRC staff finds that the licensee's evaluation is bounded by the licensing code of record and the original design basis; and, therefore, the CRDMs are acceptable for the power uprate.

3.5.5 Reactor Coolant Pumps (RCPs)

The licensee reviewed the existing design-basis analyses of the BVPS-1 and 2 RCPs to determine the impact of the revised design conditions. The RCPs are affected by the RCS pressure, SG outlet temperature, and NSSS design transients. After the power uprate, the RCS pressure and NSSS design transients remain unchanged. The most limiting design parameter (the steam generator outlet temperature, as provided in Table 3-1 of Enclosure 1 to the January 18, 2001, letter) decreases slightly from 541.8 °F to 541.3 °F. This temperature is lower than the design-basis temperature and, therefore, represents a less severe condition. As a result, the licensee concluded that the existing stress analyses for the RCPs at BVPS-1 and 2 are bounding for the 1.4-percent power uprate.

On the basis of its review of the information provided by the licensee, the NRC staff concurs with the licensee's conclusions that the RCPs, when operating at the proposed conditions of the 1.4-percent power uprate, will remain in compliance with the requirements of the codes and standards under which BVPS-1 and 2 were originally licensed. Therefore, the NRC staff finds the RCPs acceptable for the power uprate.

3.5.6 Steam Generators

The revised NSSS design parameters which were used as the basis for the proposed power uprate condition in comparison to the existing structural and fatigue analyses of the steam generators at BVPS-1 and 2. The licensee indicated that the evaluation was done based on the Model 51 steam generators (SGs) stress reports. The licensee indicated that the comparison of key parameters shows that the operating conditions with the proposed 1.4-percent power uprate and 30-percent steam generator tube plugging are slightly higher than the parameters for the current design-basis analyses at BVPS-1 and 2. Scale factors were developed based upon the increase in the NSSS design parameters and were applied to the baseline analysis results to develop revised stresses and fatigue usage. As a result, the licensee concluded that the Model 51 SGs at BVPS-1 and 2 will continue to meet the requirements of the ASME Code, Section III, 1965 Edition, with addenda through winter 1966 for BVPS-1 and the 1971 Edition for BVPS-2. The NRC staff finds that the licensee's evaluation is conservative, and that the structural integrity of the SGs is acceptable for the power uprate.

The licensee also evaluated the impact of the revised design conditions associated with the 1.4-percent power uprate on the potential for the flow-induced vibration and the design-basis fatigue analysis for the U-bend tubes. This evaluation focused on the most susceptible SG tubes in the plant. The licensee found that based on the uprated conditions of a minimum steam pressure of 760 psi, one tube in BVPS-1 and two tubes in BVPS-2 will require plugging after a cycle of power uprate operation due to flow-induced vibration (the CUFs for these tubes are greater than 1.0 for the 40-year life). The subject tube for BVPS-1 is row 10, column 53, in

SG “C,” and the tubes for BVPS-2 are row 8, column 60, in SG “A” and row 8, column 69, in SG “C.” The licensee indicated that these three tubes are considered adequate for at least one cycle following the implementation of the power uprate. The licensee has committed to plug these tubes (documented as Commitment No. 11 in Attachment C to FENOC’s letter dated January 18, 2001). The NRC staff finds this commitment to be a conservative and acceptable measure.

On the basis of its review, the NRC staff concurs with the licensee’s conclusion that the existing Model 51 SGs at BVPS-1 and 2 will maintain their structural and pressure boundary integrity and remain in compliance with the ASME Code of record specified in the UFSAR and, therefore, are acceptable for the proposed 1.4-percent power uprate.

3.5.6.1 SG Hardware Changes and Additional Evaluation

3.5.6.1.1 SG Tube Mechanical Plug

The mechanical plug is anchored in the SG tube with adequate friction to prevent it from dislodging and to maintain adequate leakage resistance for the limiting steady-state and transient loadings. The licensee assessed the impact of the changes to the thermal transients for the power uprate on the Westinghouse mechanical plugs, including the “short” Alloy 600 and Alloy 690 versions and the “long” Alloy 690 version. For the Westinghouse plugs, the licensee determined that the plug stresses are less than the allowable limit, which indicates that all primary stress limits are satisfied for the plug shell wall between the top land and the plug end cap. The plug continues to meet the Class 1 fatigue exemption requirements per Article N-415.1 of the 1966 Edition of Section III of the ASME Code, which is equivalent to NB-3222.4 of the 1989 Edition of the ASME Code. The licensee concluded that the usage factor of the plug will remain within the applicable ASME Code limit of 1.0 because the fatigue exemption requirements are satisfied. The licensee also evaluated the rolled Alloy 600 and 690 tube plugs manufactured by Framatome Technologies. The licensee concluded that the Framatone plugs will continue to perform their intended function adequately and will meet Section III of the ASME Code. The NRC staff agrees with the licensee’s conclusions regarding the integrity of the mechanical plugs and, therefore, finds the mechanical plugs acceptable for the power uprate.

3.5.6.1.2 SG Laser-Welded Sleeves

The licensee evaluated the laser-welded sleeves assuming that 30 percent of the SG tubes are plugged. The applicable design transients for the uprated conditions are unchanged from the zero-percent tube plugging. The cumulative fatigue usage factor of the sleeves remains less than unity and, therefore, is acceptable. The licensee concluded that structural limits for pressure, stress-range, and fatigue for the laser-welded sleeves continue to meet Section III of the ASME Code under the power-uprated conditions. The NRC staff agrees with the licensee’s conclusion and finds the integrity of the laser-welded sleeves acceptable for the power uprate.

3.5.6.2 Inspection Program and Tube Repair Criteria

The TSs for BVPS-1 and 2 require that the tubes be repaired when degradation becomes greater than 40-percent through wall. The licensee stated that the 40-percent repair criterion applies only to anti-vibration bar (AVB) wear and cold leg thinning degradation. Tubes having outside diameter stress corrosion cracking (ODSCC) at tube support plate intersections are

repaired in accordance with the voltage-based alternate repair criteria in the TSs. All other degradation mechanisms are repaired on detection. The licensee's repair approach is consistent with the general and acceptable practices in the nuclear industry.

The licensee reported that AVB wear rates are small for BVPS-1 and 2. No tubes were reported with AVB wear depths greater than 40-percent through wall for either of the last two inspections performed at both units. Experience with power uprates at other plants has shown that a significant increase in steam flow (>5-percent) and a significant decrease in steam pressure (>100 psi) may affect the flow-induced tube vibration and result in increased AVB wear. However, the 1.4-percent power uprate slightly increases the steam flow rate by about 1.6 percent and slightly decreases the steam pressure. The licensee concluded that the 1.4-percent uprate will have a negligible impact on the projected AVB wear rate and will not significantly impact future tube wear at the AVBs. In the unlikely event that the AVB wear rate does increase, a sufficient safety margin remains in the tube to allow the detection of the AVB wear and to repair degraded tubes under the existing inspection and/or assessment program.

Cold leg thinning is related to flow stagnation conditions in the region of the lower peripheral tube support plate regions and localized chemistry conditions. The licensee reported that, to date, cold leg thinning indications have been reported at BVPS-1 but not at BVPS-2. The licensee indicated that the increase in operating temperature and changes in hydraulic conditions due to the uprate are minimal. Changes in localized crevice chemistry will also be insignificant. Considering these factors, the 1.4-percent uprate is expected to have a negligible impact on the initiation or growth of cold leg thinning. In addition, the initiation and growth of this degradation mechanism are included in the SG assessment and inspection program, and the licensee will evaluate any changes in the rates in the course of its condition monitoring and operational assessments. The licensee concluded that the power uprate should have minimal or no effect on either AVB wear growth or cold leg thinning.

Besides AVB wear and cold leg thinning, both units have experienced ODSCC at tube support plate intersections, stress corrosion cracking in sludge piles, primary water stress corrosion cracking (PWSCC) and ODSCC in the expansion transition regions, PWSCC in the small-radius U-bends, and wear from loose parts.

The hot leg temperature (T_{hot}) affects the initiation and growth of stress corrosion cracking degradation mechanisms. The increase in T_{hot} as a result of the power uprate is expected to be relatively insignificant (< 1.0^0 F) and should, therefore, have a negligible impact on the initiation or growth of stress corrosion cracking. The licensee will include the T_{hot} increase in its crack growth rate analyses and will consider the results in its SG inspection program.

Chemistry conditions can also influence the initiation and growth of stress corrosion cracking. The licensee will not change the primary or secondary water chemistry regimes associated with the 1.4-percent uprate. Additionally, the proposed uprate will have no significant effects on primary or secondary water chemistry conditions that would adversely affect the degradation of the SG tubes. Furthermore, changes in hydraulic conditions associated with the 1.4-percent uprate are not expected to have an adverse effect on loose parts wear. The licensee will revalidate existing analyses associated with loose parts wear for the 1.4-percent uprate conditions.

With respect to the inspection program, the licensee will follow the recommended inspections in NRC Generic Letter (GL) 95-05, “Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,” as well as accepted industry guidelines. GL 95-05 provides guidance on the implementation of voltage-based alternate repair criteria for ODSCC at tube support plate intersections. The NRC staff has approved the voltage-based alternate repair criteria for BVPS-1 and 2. Although the licensee has not implemented the voltage-based repair criteria at BVPS-2, it has inspected the BVPS-2 tubes in accordance with GL 95-05. The only variance from this program is that for BVPS-2, all distorted signal indications at tube support plate intersections, regardless of amplitude, are inspected using a rotating coil probe. Any BVPS-2 tube support plate indication confirmed by the rotating coil probe, regardless of bobbin signal amplitude, will be repaired, until such time that the voltage-based repair criteria are implemented. For both units, the licensee inspects the full length of all tubes using a bobbin probe and the hot leg top of the tubesheet region of all tubes using a rotating coil probe.

With respect to the proposed power uprate, the licensee committed to consider the higher temperatures in crack growth rate analyses conducted as part of the SG inspection programs for BVPS-1 and 2. The licensee will expand its inspection plans and repair tubes on the basis of results of the condition monitoring and/or operational assessments. The licensee will also incorporate degradation growth rate changes into the operational assessment associated with the potential effects of the uprate. In that way, the licensee will confirm that the existing 40 percent through-wall plugging criterion for degraded SG tubes will remain adequate for the 1.4-percent uprate conditions. The NRC staff agrees with the licensee’s conclusion that the 1.4-percent power uprate will have an insignificant impact on the tube repair criteria and existing degradation mechanisms. Furthermore, the NRC staff finds that the licensee’s stated methods are acceptable as a means of accounting for the potential effects of the power uprate in the SG inspection program.

3.5.7 Pressurizer

The licensee evaluated the structural adequacy at limited locations in the pressurizer and components under the uprated conditions. The evaluation was performed by comparing the key parameters in the current BVPS-1 and 2 pressurizer stress reports against the revised design conditions in Table 3-1 of Enclosure 1 of FENOCs January 18, 2001, letter, for the proposed power uprate. These parameters include the RCS hot leg temperature (T_{hot}), the RCS cold leg temperature (T_{cold}), and the pressurizer transients. This comparison revealed that the changes in T_{hot} and T_{cold} are very small and are enveloped by the current stress analysis. The design transients are also unaffected by the uprated conditions. In addition, the pressurizer stress and fatigue analyses are not affected by the proposed power uprate condition. The licensee concluded that for plant operation at the 1.4-percent uprate conditions, the pressurizer components continue to meet the stress and fatigue analysis requirements of the ASME Code, Section III, 1965 Edition with addenda through winter 1966 for BVPS-1, and the 1971 Edition with addenda through summer 1972 for BVPS-2, which are the codes of record. The NRC staff finds that the licensee’s evaluation is bounded by the licensing codes of record and the original design basis and, therefore, concludes that the pressurizer is acceptable for the power uprate.

3.5.8 NSSS Piping and Pipe Supports

To evaluate the NSSS piping and supports, the licensee reviewed the design-basis analysis against the uprated power conditions, with regard to the design system parameters, transients and dynamic LOCA loads. The licensee performed this evaluation for the RCS piping, primary equipment nozzles, primary equipment supports, main steam, feedwater, high-pressure heater drains, component cooling water (CCW), spent fuel pool cooling system piping and the pressurizer surge line piping systems. The methods, criteria, and requirements used in the existing design-basis analysis for BVPS-1 and 2 are applicable for the power uprate evaluation.

The RCS pressure remains unchanged for the proposed core power uprate, while T_{hot} is projected to be slightly greater than at the current-rated power level and T_{cold} will be slightly less than at the current power level. The licensee indicated that there is sufficient margin in the existing analysis for stresses associated with the temperature changes defined in Table 3-1 of Enclosure 1 to FENOC's January 18, 2001, letter for the proposed power uprate. Moreover, the slight increase in T_{hot} improves the surge line stratification condition since it reduces the temperature differential between the pressurizer and the hot leg.

In evaluating the LOCA condition, the licensee determined that the current LOCA hydraulic forcing functions are bounding for the uprated power condition for BVPS-1 and 2. Therefore, the loads on the reactor coolant loop piping and nozzles in the existing analyses are bounding for the proposed power uprate. The licensee indicated that the design transients used in the evaluation of the RCS piping systems and equipment nozzles are unchanged for the BVPS-1 and 2 power uprate. As a result, the licensee concluded that all piping systems will remain acceptable and will continue to satisfy the design-basis requirements in accordance with applicable design-basis criteria, when considering the temperature, pressure, and flow rate effects resulting from the power uprate conditions. Specifically, BVPS-1 piping and related support systems will remain within allowable stress limits in accordance with American National Standards Institute (ANSI) B31.1, 1967 Edition, including the 1971 Addenda. Similarly, BVPS-2 piping and related support systems remain within allowable stress limits in accordance with ASME Code, Section III, 1971 Edition with addenda through winter 1972 for Class 1, 2, and 3 piping, and ANSI B31.1, 1967 Edition with addenda through June 30, 1972, for Class 4 piping.

On the basis of its review of the licensee's submittal, the NRC staff finds that the maximum calculated stresses and CUFs provided in Tables 1 and 2 of Enclosure 1 of the licensee's January 18, 2001, letter, are less than the limits allowed by the Code. The NRC staff concurs with the licensee's conclusion that the existing NSSS piping and supports, the primary equipment nozzles, the primary equipment supports, and the auxiliary lines connecting to the primary loop piping will remain in compliance with the requirements of the design-basis criteria, as defined in the BVPS-1 and 2 UFSARs, and are therefore, acceptable for the power uprate.

3.5.9 Fuel Assembly

FENOC evaluated the impact of the 1.4-percent uprate on the structural integrity of the 17x17 Vantage 5 Hybrid (V5H) fuel design fuel assemblies for BVPS-1 and 2. This evaluation revealed that the uprate will have no effect on the core plate motions used in the seismic and LOCA evaluations. FENOC also determined that the uprate conditions do not increase the operating and transient loads to the extent that they adversely affect the fuel assembly functional requirements. On that basis, the licensee concluded that the fuel assembly structural

integrity is not affected and the seismic and LOCA evaluations of the 17x17 V5H fuel design remain valid. Having reviewed this information, the NRC staff concludes that the licensee's assessment is conservative and that the fuel used for the power uprate will perform within the previously established acceptance criteria. The NRC staff, therefore, finds the structural integrity of the fuel assemblies acceptable for the power uprate.

3.6 NSSS/Balance of Plant (BOP) Fluid Systems Interface

3.6.1 Main Steam System

The licensee evaluated the effects of the proposed 1.4-percent power uprate on the main steam system, including the main steam safety valves, atmospheric steam dump valves, and residual heat control valves. The licensee concluded that the components are adequately sized for the power uprate and that the current design basis is still valid.

Based on the information provided by the licensee and the experience gained from prior NRC staff reviews of power uprate applications for similar PWR plants, the NRC staff finds the licensee’s evaluation appropriate and conservative. The NRC staff concurs with the licensee that operations at the proposed 1.4-percent uprate power level will have little impact on the operation of the main steam system and its associated components. Therefore, the NRC staff finds the main steam system acceptable for the power uprate.

3.6.2 Condenser Steam Dump System

The licensee evaluated the effects of the proposed 1.4-percent power uprate on the condenser steam dump system. The licensee determined that no hardware or operational modifications are required and that the current design basis remains valid.

Based on the information provided by the licensee and the experience gained from prior NRC staff reviews of power uprate applications for similar PWR plants, the NRC staff finds the licensee’s evaluation appropriate and conservative. The NRC staff finds that operations at the proposed 1.4-percent uprate power level will have little impact on the operation of the condenser steam dump system. Therefore, the NRC staff finds the condenser steam dump system acceptable for the power uprate.

3.6.3 Condensate and Feedwater System

The licensee evaluated the effects of the 1.4-percent power uprate on the condensate and feedwater system, which automatically maintains steam generator water levels during steady-state and transient operations. The major components of the system are main feedwater control valves, the main feedwater isolation valves, and the condensate and feedwater pumps. The revised design conditions will impact feedwater volumetric flow and system pressure drop. The licensee concluded that the components are adequately sized and that the current design basis remains valid. The licensee determined that the 1.4-percent power uprate will have no impact on the condensate and feedwater system.

The NRC staff concurs with the licensee’s conclusion. Based on the information provided by the licensee and the experience gained from prior NRC staff reviews of power uprate

applications for similar PWR plants, the NRC staff finds that the licensee's evaluation is conservative and that the condensate and feedwater system is acceptable for the power uprate.

3.6.4 Auxiliary Feedwater System (AFWS)

The licensee evaluated the effects of the 1.4-percent power uprate on the AFWS and the primary plant distilled water storage tank (PPDWST). The AFWS is normally aligned to take suction from the PPDWST. In addition to the AFWS being able to meet the minimum feedwater flow requirement, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition. The minimum PPDWST usable inventory requirement is based on 102 percent of the current rated power, which bounds the proposed power uprate. The licensee has determined that no hardware or operational modifications are required and that the current design basis remains valid. The licensee concluded that the 1.4-percent power uprate will have little or no impact on the operation of the AFWS and the PPDWST.

Based on the information provided by the licensee and the experience gained from prior NRC staff reviews of power uprate applications for similar PWR plants, the NRC staff finds the licensee's evaluation conservative. The NRC staff finds that operations at the proposed 1.4-percent uprate power level will have little impact on the operation of the AFWS. Therefore, the NRC staff finds the AFWS acceptable for the power uprate.

3.6.5 Steam Generator Blowdown (SGBD) System

The function of the SGBD system is to control chemical composition and buildup of solids in the SG shell-side water. The blowdown flow rates are determined by water chemistry, and by the tube-sheet sweep required for controlling buildup of solids. The rate at which dissolved solids are introduced into the secondary water of the SG depends on condenser leakage, quality of secondary makeup water, and the amount of corrosion products generated by flow accelerated corrosion (FAC). Only the generation of corrosion products by the FAC could be affected by the power uprate because it increases with increasing flow, which is expected to occur after the power uprate. However, the licensee determined that this flow increase is minimal and the change in generation of corrosion products is insignificant. Therefore, the chemistry environment in the shell-side of the SGs will not change as a result of the power uprate.

The flow control valves in the blowdown system are designed for a specified range of inlet pressures between no load and full load plant operation. The licensee determined that these pressures will not change and there will be no need to modify these valves. The licensee concluded, therefore, that the power uprate will not produce any effect which would downgrade the operation of the SGBD system operation.

Based on the information provided by the licensee, the NRC staff finds the licensee's evaluation appropriate and conservative. The NRC staff finds that operations at the proposed 1.4-percent uprate power level will have little impact on the operation of the SGBD system. Consequently, the NRC staff finds the SGBD system acceptable for the power uprate.

3.7 Balance of Plant (BOP) Systems

To evaluate the adequacy of the BOP systems, the licensee compared the existing design-basis parameters with the core power uprate conditions. The safety-related BOP piping systems evaluated were the reactor/primary component cooling water, river/service water, and containment depressurization systems. The non safety-related systems evaluated were the extraction steam, heater drains, circulating water, and turbine/secondary component cooling water systems. These piping systems, together with the RCS piping and supports, were evaluated for the effects resulting from the revised NSSS parameters (RCS temperatures, steam temperature and steam flow rate) and the heat balance at 2689 MWt. As a result, the licensee concluded that the existing design-basis analyses for the BOP piping systems are acceptable for the uprated power level of 2689 MWt at BVPS-1 and 2.

Based on the information provided by the licensee and the experience gained from the NRC staff reviews of power uprate applications for similar PWR plants, the NRC staff finds the licensee's evaluation conservative. The NRC staff concurs with the licensee and finds that operations at the proposed 1.4-percent uprate power level will have little impact on the operation of these BOP systems. Therefore, the NRC staff finds these systems acceptable for the power uprate.

3.8 Electrical Systems

The BVPS-1 and 2 receive shutdown power from two physically independent and redundant offsite power sources of the 138 kilo-volt (kV) switchyard system. The isolated phase bus is designed to deliver power from the main generator terminals to the main power transformers and is connected "wye" on the high-voltage side and "delta" on the low-voltage side. The station service power is supplied from either the main generator, the switchyard, or a combination of both. On failure of the preferred source, automatic throw-over capability is provided to the alternate source to ensure continuous power to the equipment. The onsite power system, which consists of the onsite alternating current (ac) power system, 125 volts (V)-direct current (dc) power system, and the 120 V-ac vital bus system, provides power to the vital station auxiliaries if a normal source of power is not available. The fast-starting emergency diesel generators provide the source of ac power for the ac onsite power system. The electrical distribution system has been previously evaluated to conform to 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power System."

3.8.1 Plant Electrical Systems (ac and dc)

The licensee reviewed the electrical distribution systems to identify the major items that may be affected by the proposed power uprate conditions. In particluar, the licensee evaluated the following systems and components:

- Main unit generator
- Isolated phase bus duct
- Switchyard equipment
- Main power transformers (MPTs)
- Unit station service transformers (USSTs)
- System station service transformers (SSSTs)
- Non-segregated phase bus ducts

- Large loads and cables
- Emergency diesel generators
- Protective relay settings

The turbine manufacturer, Siemens Westinghouse, evaluated the capability of the turbine generators at the proposed uprate power conditions. The review included the throttles, high-pressure and low-pressure turbines, generators and exciters, and associated auxiliary equipment. All turbine generator components were determined to have a sufficient margin to enable operation at the uprated conditions without requiring equipment modifications. Based on the revised heat balances provided by the turbine generator manufacturer, the generators will deliver gross output power of approximately 898 megawatts electric (MWe) for BVPS-1 and 908 MWe for BVPS-2 with the power uprate. This increased power output is well within the generator nameplate rating of 1026 MVA at 0.9 power factor. Therefore, the generator can operate at the uprated power level with no modifications.

The isolated phase duct, main power transformers, and associated cooling equipment are designed to accept the maximum generator output and, therefore, will continue to support plant operations at the uprated power conditions.

The switchyard equipment (345 kV switches and breakers) are rated at 2000 amperes. Due to the power uprate, the main generator output current is expected to be approximately 1700 amperes at its nameplate rating of 1026 MVA and 345 kV. Therefore, the additional load is well within the rating of the switchyard equipment without the need for any hardware modifications.

The bus loading summaries for connected 4.16 kV switchgear under the uprate conditions remain less than the USSTs and SSSTs design ratings. The cooling equipment associated with the USSTs will also support continuous operation under the uprated power conditions with no modifications.

The non-segregated phase duct connects the USSTs and SSSTs to their respective 4.16 kV switchgear. The non-segregated phase bus duct runs have a continuous rating of 2500 amperes per phase at 4.16 kV. The bus loading summaries for connected 4.16 kV switchgear under the uprate conditions confirm that the non-segregated phase ducts are adequate.

System reviews confirmed that a few of the large medium-voltage motors on the nonsafety-related 4.16 kV switchgear will experience a load change under the uprated power conditions. Load flow analyses performed for 4.16 kV bus loads under the uprated power conditions verify acceptable loading. Therefore, the large station auxiliary loads and associated cables are considered adequate as installed, and the motors will continue to satisfactorily perform their intended functions.

There is no change to the safety-related loads at uprate conditions and, therefore, the emergency diesel generators will not be impacted by the proposed uprate and will remain capable of performing their safety-related functions during a loss-of-offsite power (LOOP)/LOCA at the uprate conditions.

The licensee further states that all other equipment and components, including station protective schemes and setpoints, will continue to support safe and reliable plant operation

under the proposed power uprate conditions. Bus voltage and fault current values at different levels of the station auxiliary electrical distribution systems will remain within acceptable limits under uprate. There are no impacts to the dc power system voltage or short circuit current levels. The NRC staff finds the ac and dc systems acceptable for the power uprate.

3.8.2 Grid Stability

Under the proposed power uprate conditions, there is no change in the engineered safety feature loads, and bus voltage values at different levels of the station auxiliary distribution systems are bounded by the existing load flow and voltage profile analysis. The 1.4-percent increase in power generated into the 345 kV system has no significant impact on the 138 kV switchyard system and the ability of the units to safely shutdown. In 1996 and 1997, the licensee performed a grid stability study that indicated that the transmission system will remain stable under worst-case postulated contingencies, and the calculated voltages at BVPS-1 and 2 345 kV and 138 kV buses remain acceptable. The licensee is currently, updating the grid stability study to ensure that the model reflects system changes that have occurred since 1997. The new study will incorporate the 1.4-percent power uprate to identify any stability issues that may require resolution. This new study will be completed before the licensee increases power above 2652 MWt (this is documented as Commitment No. 9 in Attachment C to FENOC's letter dated January 18, 2001).

3.8.3 Equipment Qualification

The licensee reviewed the qualified life of Class 1E and other post-monitoring instrumentation to identify any impact attributable to environmental conditions that may result from operating the plant at uprated power. The environmental radiation levels for both normal operation and accident conditions were originally developed using assumed power levels that envelope the uprate conditions. For the accident contribution, margins were incorporated into the equipment specifications that met or exceeded the requirements of IEEE 323-1974. Therefore, on the basis of these considerations, the NRC staff finds that the equipment qualification remains acceptable for operation at the uprated power level.

3.8.4 Summary

The NRC staff evaluated the effect of the proposed power uprate on the necessary electrical systems and environmental qualification of electrical components. The results of this evaluation show that the licensee's evaluation is conservative and the proposed increase in core thermal power would have negligible impact on the plant's ac and dc electrical systems and components, grid stability (contingent on grid stability reevaluation), or environmental qualification consistent with General Design Criterion (GDC) 17. Therefore, the NRC staff finds the ac and dc electrical systems acceptable for the power uprate.

3.9 Nuclear Steam Supply System (NSSS) Accident Evaluation

3.9.1 Steam Generator Tube Rupture (SGTR) Evaluation

The licensing basis SGTR analysis for BVPS-1 was performed using a simplified mass and energy balance method. The input parameters that change as a result of the uprate include power, hot leg temperature (T_{hot}), cold leg temperature (T_{cold}), steam temperature (T_{steam}), and

steam pressure (P_{steam}). An increase in reactor power would slightly change these parameters, resulting in an increase in steam release due to a small increase in system energy. However, the methodology used in the current licensing-basis analysis includes a 4.5-percent margin in reactor power for the calculation of the feedwater flows and steam releases. The analyzed 104.5-percent reactor power bounds the power uprate of 1.4 percent. Therefore, the NRC staff finds the current analysis acceptable for the power uprate.

The licensing-basis SGTR analysis for BVPS-2 was performed using the LOFTTR2 methodology. This methodology utilizes the LOFTTR2 computer code to model the SGTR thermal and hydraulic characteristics and the margin to SG overfill. The power rating for the BVPS-2 SGTR analysis is 102 percent (2705 Mwt) which bounds the power uprate of 1.4 percent with a 0.6-percent uncertainty. Therefore, the NRC staff finds the current analysis acceptable for the power uprate.

3.9.2 Loss of Coolant Accident (LOCA)-Related Analyses

3.9.2.1 Large-Break LOCA (LBLOCA) and Small-Break LOCA (SBLOCA)

The current BVPS-1 and 2 LBLOCA and SBLOCA analyses assume a maximum steady-state reactor core power of 102 percent (2705 MWt) of the current licensed power level (2652 MWt). FENOC proposed to continue using the present LBLOCA and SBLOCA analyses at 2705 MWt as the licensing-basis analyses for BVPS-1 and 2. The proposal is based on its use of the Caldon LEFM technology to reduce the power measurement uncertainty to 0.6 percent from the previously assumed 2.0 percent. The remaining 1.4-percent margin is available for increasing the licensed power without changing the LOCA analyses initial power assumptions.

In its letter dated June 29, 2001, the licensee showed that the LBLOCA and SBLOCA analysis methodologies presently approved for BVPS-1 and 2 continue to apply by providing a statement that the licensee and its vendor(s) have ongoing processes that ensure that LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as-operated plant values for those parameters.

The licensee concluded that the uprated LOCA analysis input initial power assumption is consistent with 10 CFR Part 50, Appendix K, Section 1.A, “Sources of Heat During a LOCA.” FENOC also demonstrated that the LBLOCA and SBLOCA analysis methodologies presently approved for BVPS-1 and 2 continue to apply. Based on the information provided by the licensee, the NRC staff finds that the LOCA analyses presently approved for BVPS-1 and 2 bound the uprated conditions and, therefore, are acceptable.

3.9.2.2 Post-LOCA Long-Term Core Cooling (LTCC)

The LTCC requirements following a LOCA are established by 10 CFR 50.46(b)(5), “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors - Long Term Cooling.” To satisfy this requirement, an adequate water volume and boron concentration must be provided to ensure that the reactor core remains subcritical assuming that all of the control rods are out. The BVPS-1 and 2 current licensing position is that the reactor remains shut down by borated emergency core cooling system (ECCS) water residing in the RCS sump following the LOCA. The water volumes and boric acid concentrations for the ECCS water supply were originally designed for the LBLOCA as the limiting case. Since the LBLOCA was

analyzed at a power level of 2705 MWt, the water volumes and boric acid concentrations of the ECCS water supply remain acceptable for the power uprate to 2689 MWt.

3.9.2.3 Hot Leg Switchover

Long-term cooling requirements following a LOCA are established by 10 CFR 50.46(b)(5). One aspect of long-term cooling is to ensure boric acid (H_3BO_3) accumulation will not prevent core cooling. Consequently, the NRC staff assessed the licensee's evaluation model (EM) for analysis of boric acid accumulation and determination of the time available for switchover to hot leg injection following a LOCA. The NRC staff understands the licensee's EM is based on the following assumptions:

1. All H_3BO_3 entering the core/upper plenum volume of the reactor vessel remains in that volume prior to initiation of hot leg injection.

2. Water in the core/upper plenum volume is well mixed by the boiling process.

3. The upper plenum collapsed water level is at the level of the bottom of the hot leg flow area at the reactor vessel.

4. The bottom of the well mixed core/upper plenum volume is at the level of the bottom of the active fuel region. (The lower plenum volume and the volume between the lower core plate and the bottom of the active fuel region are not included in the core/upper plenum volume.)

5. Water in the downcomer, lower plenum, and at the top surface of the upper plenum water is at 212 °F.

6. The H_3BO_3 concentration limit is the experimentally determined H_3BO_3 saturation concentration with a 4 weight percent uncertainty factor. There is no allowance for increase in H_3BO_3 solubility due to other solutes such as sodium hydroxide.

7. The decay heat generation rate is 1.0 times the draft 1971 ANS Standard based on three core regions with operating times of 8000, 16000, and 24000 hours, respectively.

8. Decay heat generation includes a suitable multiplier to address instrumentation uncertainty as identified by Section I.A of Appendix K to 10 CFR, Part 50. This is either a 1.02 power multiplier or a value that has been acceptably demonstrated to account for uncertainties due to power level instrumentation error.

9. The containment contains the maximum deliverable water volumes at the maximum allowable H_3BO_3 concentration.

10. The spray additive tank is credited as an H_3BO_3 dilution source using a minimum volume assumption. This results in approximately a 5-minute increase in the maximum allowable time to switchover to hot leg injection when compared to excluding the volume.

11. The calculation neglects any elevation of boiling temperature due to concentration of H_3BO_3 in the core or any backpressure from containment.

12. The barrel/baffle region volume is neglected.

Generally, the NRC staff will accept inclusion of the volume between the bottom of the active fuel and the top of the core support plate as part of the core/upper plenum mixing volume; however, NRC staff guidance does not recommend including boric acid dilution sources. Assumption 4 excludes the above volume and assumption 10 includes a dilution source. One assumption is conservative; the other is not. This combination of compensating assumptions is acceptable.

Assumption 7 (decay heat generation rate of 1.0 times the ANS Standard for a finite operating time) is inconsistent with the 10 CFR Part 50, Appendix K, Section I.A.4 specification of 1.2 times the ANS Standard for an infinite operating time. However, the licensee's assumed decay heat generation rate is consistent with that described in a letter dated April 1, 1975, from C. L. Case, Manager, Safeguards Engineering, Westinghouse Corporation Power Systems, CLC-NS-309, to T. M. Novak, Chief, Reactor Systems Branch, NRC, which the NRC staff understands represents the approach that is generally used by licensees of Westinghouse-designed nuclear power systems.

There is a low probability that conditions leading to significant H_3BO_3 accumulation will be encountered. Further, a partial quantification of approved EM conservatisms shows that 1.2 times the values for infinite operating time in the 1971 ANS Standard represents less than one third of the total conservatisms. Consequently, the NRC staff does not consider the inconsistency to be a safety-significant difference and is assessing this inconsistency on a generic basis. Therefore, since the licensee uses a methodology that has historically been applied to such analyses, with the above noted exceptions that do not significantly impact its predictions, the NRC staff will continue to accept the licensee's analysis pending completion of an NRC generic assessment.

3.9.4 Non-LOCA/Transient Analyses

For the non-LOCA transients, the non-departure from nucleate boiling (DNB) events were analyzed at a power level of 2705 MWt. This value is 102 percent of the current power level, which bounds the uprated power level and associated uncertainty. However, for non-LOCA DNB events, the licensee performed analyses as described in its letter dated December 27, 2000. The NRC staff approved these analyses on July 20, 2001, in Amendment Nos. 239 and 120 to FOL Nos. DPR-66 and NPF-73 for BVPS-1 and 2, respectively. These amendments approved, among other things, a modification to the design and safety analysis DNB ratio (DNBR) limits based upon the Westinghouse revised thermal design procedure (RTDP) methodology. The review concluded that the non-LOCA RTDP analyses were performed using an approved methodology at conditions consistent with the uprated power level.

In addition, the licensee analyzed the non-LOCA transients at the uprated conditions as presented in FENOC's letter dated June 29, 2001. These analyses showed that the minimum DNBR for all transients remains above the safety analysis DNBR limits. Based upon the results of the calculations, the acceptance criteria continued to be met for the non-LOCA transients. Therefore, the NRC staff finds that the BVPS-1 and 2 non-LOCA transient analyses are acceptable for the uprated power condition.

3.9.5 Revised Thermal Design Procedures (RTDP) Uncertainties

3.9.5.1 Power Calorimetric

The current plant safety analyses assume a total power calorimetric uncertainty of 2.0 percent. However, the licensee plans to use the Caldon LEFM✔™ and LEFM CheckPlus™ systems for feedwater mass flow measurements at BVPS-1 and 2, respectively. These systems reduce the uncertainty associated with the power calorimetric by more accurately measuring the secondary-side feedwater mass flow. The reduced feedwater flow measurement error, in combination with the remaining uncertainty components related to the power calorimetric, results in a total 95/95 power measurement uncertainty of ±0.6 percent of rated thermal power.

The NRC staff reviewed and approved a ±0.6-percent reactor power uncertainty for use with the Caldon LEFM✔™ system in other licensing applications. Uncertainties associated with the Caldon LEFM CheckPlus™ system were approved by the NRC staff in License Amendment Nos. 239 and 120, dated July 20, 2001, for BVPS-1 and 2, respectively. The NRC staff found that the Caldon LEFM CheckPlus™ system was at least as accurate as the Caldon LEFM✔™ system. Consequently, total power uncertainty associated with utilization of the Caldon LEFM CheckPlus™ system at BVPS-2 was found be ±0.6 percent.

3.9.5.2 T_{avg} Rod Control, Pressurizer Pressure Control and RCS Flow Calorimetric

For T_{avg} rod control, the nominal full-power turbine impulse pressure is the T_{avg} reference. Since the power uprate is only a minor change in the turbine impulse pressure, the licensee claims that the power uprate does not affect the turbine final calculated T_{avg} rod control uncertainties. Similarly, FENOC states that the pressurizer pressure control system uncertainties are not affected by the power uprate conditions. Also, for the RCS flow calorimetric, the licensee claims that the changes in plant parameters due to the uprated conditions do not change the final calculated uncertainties, as referenced in WCAP-15264, Revision 3, and WCAP-15265, Revision 3, which were submitted in support of the review of License Amendment Nos. 239 and 120, dated July 20, 2001, for BVPS-1 and 2, respectively.

The NRC reviewed and approved the uncertainties for T_{avg}, pressurizer pressure, and RCS flow in the documents that were submitted in support of Amendment Nos. 239 and 120, dated July 20, 2001, for BVPS-1 and 2, respectively. The NRC staff found these uncertainties acceptable. The NRC staff's evaluation, as documented in Amendment Nos. 239 and 120, dated July 20, 2001, continue to be valid. Therefore, the NRC staff find the uncertainties associated with the T_{avg} rod control, pressurizer pressure control, and RCS flow calorimetric acceptable for the power uprate.

3.9.5.3 RTS/ESFAS Uncertainties

The reactor trip system (RTS) and engineered safety features actuation system (ESFAS) setpoint uncertainties are defined in Westinghouse WCAP-11419, Revision 2, and WCAP-11366, Revision 4. These topical reports were reviewed and approved as part of Amendment Nos. 239 and 120, dated July 20, 2001, for BVPS-1 and 2, respectively. The RTS/ESFAS functions that could be affected by the power uprate include loss of flow (low

reactor coolant flow), steam generator water level, overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT).

3.9.5.3.1 RCS Loss of Flow (Low Reactor Coolant Flow)

As stated in Section 3.9.5.2 of this SE, the power uprate plant parameter changes do not affect RCS flow calorimetric uncertainties. Since these uncertainties are not affected, the licensee states that the low reactor coolant flow trip uncertainties do not need to be modified. The NRC staff agrees and finds the uncertainties associated with RCS flow acceptable.

3.9.5.3.2 Steam Generator (SG) Water Level

For the SG water level uncertainties, FENOC stated that the small changes in nominal steam pressure, feedwater temperature, recirculation ratio, and reference leg temperature effects have negligible impact. Therefore, the licensee states that the uncertainties for the SG water level trips also do not need to be modified. Based on the information provided, the NRC staff agrees and finds the uncertainty associated with the SG water level acceptable.

3.9.5.3.3 OTΔT and OPΔT

In reference to the OTΔT and OPΔT trips, FENOC claims that the uprate will not necessitate changes to the trip uncertainties. Upon review of the uncertainty values in the accident analyses, the NRC staff determined that the changes to the OTΔT and OPΔT trip uncertainties are bounded by those of the accident analyses. In addition, the uncertainty values were reviewed and approved by Amendment Nos. 239 and 120, dated July 20, 2001, for BVPS-1 and 2, respectively. The NRC staff found the uncertainties associated with the OTΔT and OPΔT functions acceptable.

3.10 Containment/BOP Accident Evaluations

3.10.1 Containment Analyses

The licensee evaluated the short-term LOCA with respect to the proposed 1.4-percent power uprate. Mass and energy release calculations were performed to support the reactor cavity and containment subcompartment pressurization analyses. The licensee determined that the design of the reactor cavity and containment subcompartments remains acceptable at the uprate power level.

The licensee also evaluated the long-term LOCA with respect to the proposed 1.4-percent power uprate. The current BVPS-1 and 2 LOCA mass and energy release analysis is based on a power level of 2713 MWt, which bounds the proposed power uprate. The BVPS-2 analysis was performed at two power levels with the analysis for the early stage of the accident being based on 2811 MWt and the later accident stages analysis being based on 2713 MWt. Both power levels bound the proposed uprate power.

The current main steam line break analysis for both units assumes a power level of 2660 MWt plus a 2-percent power calorimeteric uncertainty for breaks inside and outside of containment. The use of the Caldon LEFM systems reduces the maximum calorimeteric uncertainty to 0.6

percent. Operation at 101.4 percent of current power with a 0.6-percent uncertainty is bounded by the current analysis. The licensee determined that in all cases the 1.4-percent power uprate will have little impact on the containment integrity analyses.

The NRC staff finds that plant operations at the proposed 1.4-percent power uprate will have an insignificant impact on containment integrity. Furthermore, the current design-basis analyses bound the uprate conditions and are, therefore, acceptable.

3.11 Radiological Consequences

The licensee performed an assessment of the design-basis accident (DBA) radiological dose consequences for the 1.4-percent power uprate to 2689 MWt. The amendment request describes a review of the impacts of the change on post-accident shielding and accident radiological consequences. To assess the potential impact of the change, the licensee reviewed the current power levels utilized in the shielding and accident analyses and compared this value to the proposed power level of 2689 MWt with 0.6-percent uncertainty. The licensee stated that the shielding analysis was performed at a power level of 2766 MWt and the accident analysis has been reanalyzed for a reactor power of 2705 MWt. The current DBA dose consequence accident analyses were approved on March 23, 2001, as documented in Amendment Nos. 237 and 119 to FOL Nos. DPR-66 and NPF-73 for BVPS-1 and 2, respectively. FENOC further stated that these amendments demonstrate that the dose guidelines set by 10 CFR Part 100 and 10 CFR Part 50, Appendix A, General Design Criterion 19, for the site boundary and control room, respectively, are met. Since the analyzed power levels of 2766 MWt and 2705 MWt described above for the shielding and accident analyses, respectively, bound the proposed uprated power plus uncertainty, FENOC concluded that the proposed change is acceptable.

The NRC staff finds that the information provided by the licensee supports the proposed change. The decrease in reactor power measurement uncertainty (from 2 percent to 0.6 percent of the RTP) due to the installation of new flow measuring equipment effectively offsets the increase of 1.4 percent in power level (2652 to 2689 MWt). Current reactor power uncertainty is taken to be 2 percent of the RTP. The current rated thermal power of 2652 MWt has an uncertainty of approximately 53 MWt (2652 MWt x 2 percent). Therefore, the shielding and accident analyses are currently designed for at least a power level of 2705 MWt (2652 MWt + 53 MWt). The requested power increase is to 2689 MWt with a measurement uncertainty of 0.6 percent of RTP. The proposed uncertainty band is equivalent to approximately 16 MWt (2689 MWt x 0.6 percent). With this uncertainty, the maximum power with uncertainty is 2705 MWt (2689 MWt + 16 MWt). Since the proposed maximum power level with uncertainty is less than or equal to the design values for shielding (2766 MWt) and the accident analyses (2705 MWt), the current analyses bound the power uprate.

The results of the NRC staff's assessment described above were used to confirm the acceptability of the licensee's analysis methodology and conclusions. Based on the considerations above, the NRC staff finds that the licensee's analyses remain acceptable.

The NRC staff finds reasonable assurance that the postulated radiological consequences of the DBAs at BVPS-1 and 2 will continue to be less than the dose guidelines of 10 CFR Part 100 and the criteria of 10 CFR Part 50, Appendix A. Therefore, with respect to the potential

radiological dose consequences of DBAs, the proposed power uprate for BVPS-1 and 2 is acceptable.

3.12 Nuclear Fuel

3.12.1 Fuel Core Design (Reactor Core Design)

To evaluate the effects of the power uprate on the current reactor core design, a representative equilibrium fuel cycle model was developed. FENOC states that the methods and core models used in the analyses are consistent with those currently in the BVPS-1 and 2 UFSARs. They also claim that the core analyses demonstrate that the uprate does not result in changes to the current nuclear design basis documented in the UFSAR. Finally, the licensee asserts that the impact on the peaking factors, rod worths, reactivity coefficients, shutdown margin, and kinetics parameters either are well within normal cycle-to-cycle variation or are controlled by the core design and will be addressed on a cycle-specific basis consistent with the reload methodology.

The NRC staff finds that the licensee's analyses and controls are conservative and appropriate for the power uprate and, therefore, the fuel core design remains acceptable for the power uprate.

3.12.2 Core Thermal-Hydraulic Design

The licensee performed the core thermal-hydraulic analyses and evaluations at the uprated core power level of 2689 MWt and assumed that the core designs were composed of V5H fuel assemblies without intermediate flow mixers. To ensure that the core meets the DNB design basis, the licensee used the RTDP and the THINC IV computer codes. The licensee also used the WRB-1 DNB correlation for the 17x17 V5H fuel assemblies.

The NRC evaluated the design and safety analysis DNBR limits using the RTDP methodology, and found them acceptable as described in Amendment Nos. 239 and 120, dated July 20, 2001, for BVPS-1 and 2, respectively.

The NRC staff finds that the licensee performed an appropriate and conservative evaluation and, therefore, finds that the core thermal-hydraulic design is acceptable for the power uprate.

3.12.3 Fuel Rod Design

FENOC reviewed the fuel rod design analyses for BVPS-1 and 2 at the uprated conditions. They found that some of the rod design criteria were negligibly impacted by the uprate. For those criteria that were impacted the most, namely the rod internal pressure and cladding stress, the licensee found that the fuel design continues to meet the acceptance criteria.

The NRC staff finds that the licensee conservatively evaluated the impacts on the fuel rod design. Furthermore, the fuel rod design continues to meet all of the acceptance criteria requirements. Therefore, the NRC staff finds the fuel rod design acceptable for the power uprate.

3.13 Programs and Other Issues

3.13.1 Human Factors

With regard to changes in emergency and abnormal operating procedures, the licensee stated in its letters dated January 18, 2001, and May 7, 2001, that, “the 1.4-percent power uprate is not expected to have any significant effect on the manner in which the operators control the plant.” Plant procedures will require only minor changes for the power uprate.

The NRC staff finds that the licensee’s response is satisfactory because the licensee will treat plant procedure changes in a manner that is consistent with any other procedure changes, and these changes will be included in operator training accordingly.

With regard to changes to risk-important operator actions that are sensitive to the power uprate, the licensee stated in its letter dated May 7, 2001, that “there are no changes to current risk-important operator actions as a result of the power uprate.” There are also no new operator actions that are being automated as a result of the power uprate.

The NRC staff finds that the licensee’s response is satisfactory because the licensee has adequately addressed the question of operator actions that are sensitive to the power uprate.

With regard to changes to control room controls, displays, and alarms, the licensee stated in its letters dated January 18 and May 7, 2001, that “a control room audible annunciator will be provided in each unit to alarm LEFM trouble or failure.” The licensee also stated that “no other changes to controls, displays, or setpoints are required as a direct result of the power uprate.”

The NRC staff finds that the licensee’s response is satisfactory because the licensee has adequately identified the changes that will occur with regard to alarms, displays, and controls as a result of the power uprate and adequately described how these changes will be accommodated.

With respect to changes to the safety parameter display system, the licensee indicated that “the post-accident monitoring instruments, including the safety parameter display system (SPDS) were reviewed and are not affected by the proposed uprate.”

The NRC staff finds that the licensee’s response is satisfactory because the licensee has adequately demonstrated that no changes will occur with regard to the SPDS as a result of the power uprate.

With respect to changes to the operator training program and the control room simulator, the licensee indicated that “benchmarking of training simulator fidelity with the new power rating will be included at the next regularly scheduled review following the uprating. Simulator revalidation is performed in accordance with ANSI/ANS-3.5-1985.” The physical changes that affect the simulator will be implemented through approved plant change processes. Training for operators in the use of the LEFM and associated alarm response procedures will be conducted prior to plant operations at uprated conditions.

The NRC staff finds the licensee's response satisfactory because the licensee has adequately described how the changes to operator actions will be addressed and how the simulator will accommodate the changes in accordance with the requirements of ANSI/ANS-3.5-1985.

The NRC staff finds that the licensee has satisfactorily addressed these human factors areas associated with the proposed power uprate. The NRC staff further finds that the power uprate should not adversely affect simulation facility fidelity, operator performance, or operator reliability. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to human factors issues.

3.13.2 Motor-Operated-Valves (MOVs)

The licensee also reviewed the programs, components, structures, and non-NSSS system operational aspects affected by the power uprate. In its letter dated January 18, 2001, the licensee stated that there are no changes to the MOV program as a result of the 1.4-percent power uprate. The safety-related valves were not found to be impacted by the 1.4-percent power uprate and are, therefore, acceptable. The licensee evaluated its commitments relating to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," associated with the pressure locking and thermal binding of safety-related power operated gate valves that are required to operate in accordance with their intended safety function. The licensee found that the existing analysis conditions remain bounding for the 1.4-percent power uprate. The licensee also evaluated its response relating to the GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," program regarding the over-pressurization of isolated piping segments. The licensee concluded that the conditions used in the existing evaluation for GL 96-06 remain bounding for the proposed power uprate of 1.4 percent. On the basis of the information provided by the licensee, the NRC staff finds the licensee's evaluation conservative and concurs with the licensee's conclusions that the power uprate will have no adverse effects on the safety-related valves and that the conclusions from the GL 95-07, GL 96-06, and GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," programs remain valid. Therefore, the NRC staff finds the programs associated with MOVs acceptable for the power uprate.

3.13.3 Flow Accelerated Corrosion (FAC)

FAC is a mechanism that causes wall thinning of high energy pipes in the power conversion system, which may lead to their failure. Since failure of these pipes may result in undesirable challenges to the plant's safety system, the licensee has a program for predicting, inspecting, and repairing or replacing the components whose wall thinning exceeds the values required for their safe operation. The program uses the CHECWORKS code for predicting thinning of the walls in the components subjected to FAC. In its submittal, the licensee committed to upgrade the code by including any change in the input parameters caused by the power uprate. The NRC finds that the the licensee's action is adequate to ensure the integrity of the high-energy pipes.

3.13.4 Rod Ejection Event

For the rod ejection event, the BVPS-1 and 2 fuel pellet enthalpy criterion for the reactivity excursion limit is based upon 200 calories-per-gram (cal/gm) for irradiated fuel and 225 cal/gm for unirradiated fuel. The 200 cal/gm value bounds the Standard Review Plan value of 280 cal/gm and, therefore, the BVPS-1 and 2 Rod Ejection Analysis meets the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 28. For the reactor pressure criterion specified in Section 15.4.8 of the Standard Review Plan, Westinghouse has generically shown that the criterion is met as documented in Westinghouse Topical Report, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors using Spacial Kinetics Methods," January 1975, WCAP-7588, Revision 1-A. FENOC states that the generic evaluation is applicable for BVPS-1 and 2. The licensee also states that, at the uprated conditions, the fuel pellet enthalpy remains below the 200 cal/gm limit. The NRC staff agrees with FENOC's assessment of the rod ejection event and finds that the licensee's evaluation is acceptable.

3.13.5 Station Blackout (SBO)

SBO is defined in 10 CFR 50.2 as the complete loss of the preferred offsite and Class 1E onsite emergency ac power system. The licensee evaluated the SBO analysis to assess the impact of the proposed power uprate. This evaluation demonstrated that the uprate will have no impact on the ability to achieve and maintain a safe shutdown of one unit during and following an SBO event. The evaluation included heat-up analysis, equipment operability and battery capacity. Based on the information provided by the licensee, the NRC staff finds that the evaluation of a potential SBO is conservative and acceptable.

3.13.6 Anticipated Transients Without Scrams (ATWS)

BVPS-1 and 2 currently rely upon the generic ATWS analyses performed by Westinghouse, which are documented in Westinghouse letter NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," December 1979. The proposed power uprate to 2689 MWt with a 0.6-percent uncertainty is bounded by the generic analyses for a 3-loop Westinghouse plant at 2785 MWt. Therefore, the current ATWS analysis for BVPS-1 and 2 is bounded, provided that future core reloads remain within the analysis assumptions of the generic Westinghouse analyses. Therefore, the NRC staff finds the current ATWS analysis acceptable for the power uprate.

3.13.7 Fire Protection (10 CFR Part 50, Appendix R)

The licensee determined that the fire protection program is not affected by the proposed 1.4-percent power uprate. The NRC staff agrees with the licensee's evaluation and finds that the licensee's fire protection program remains acceptable for the power uprate.

3.13.8 Containment Integrity (10 CFR Part 50, Appendix J)

The licensee determined that the containment integrity program is not affected by the proposed 1.4-percent power uprate. The NRC staff agrees with the licensee's evaluation and finds that the licensee's containment integrity program remains acceptable for the power uprate.

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

With respect to the portions of the amendments related to the 1.4-percent power uprate, 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 September 13, 2001 (66 FR 47699). Accordingly, based upon the environmental assessment, the NRC staff has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

With respect to the portion of the amendment related to the revisions to BVPS-2 TS 3/4.4.9, “Pressure/Temperature Limits,” the 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 and changes 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 (66 FR 39211). Accordingly, this portion of 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 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.

Principal Contributors: D. Cullison, M. Blumberg, I. Ahmed, C. Goodman, S. Peters, W. Lyon, F. Orr, C. Wu, N. Trehan, G. Georgiev, J. Tsao, K. Parczewski, L. Lois

Date: September 24, 2001

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NOS. 50-334 AND 50-412

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos. 243 and 122 to Facility Operating License Nos. DPR-66 and NPF-73, respectively, issued to FirstEnergy Nuclear Operating Company, et. al., (the licensee) , which revised the Technical Specifications (TSs) and Operating Licenses for operation of the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) located in Shippingport, Pennsylvania. The amendment is effective as of the date of issuance.

The amendment modified the TSs and OLs to reflect an increased maximum steady-state core power level from 2652 megawatts thermal (MWt) to 2689 MWt, an increase of approximately 1.4 percent. These increases are facilitated by the utilization of the Caldon Leading Edge Flowmeter for feedwater flow measurements.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on June 19, 2001 (66 FR 32963). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the portion of the action related to the power uprate and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (66 FR 47699).

For further details with respect to the action, see (1) the application for amendment dated January 18, 2001 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML010230096), as supplemented by letters dated February 20 (ADAMS Accession No. ML010540305), April 12 (ADAMS Accession No. ML011130105), May 7 (ADAMS Accession No. ML011340076), May 18 (ADAMS Accession No. ML011440046), June 9 (3 letters) (ADAMS Accession Nos. ML011640192, ML011640189, and ML011640086), June 26 (ADAMS Accession No. ML011840215), June 29 (ADAMS Accession No. ML011870434), August 21, (ADAMS Accession No. ML012400228), and September 5, 2001 (ADAMS Accession No. ML012550393), (2) Amendment Nos. 243 and 122 to License Nos. DPR-66 and NPF-73, respectively, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, http://www.nrc.gov/NRC/ADAMS/index.html. Persons who do not have access to ADAMS or

who encounter problems in accessing the documents located in ADAMS should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, or 301-415-4737, or by e-mail at pdr@nrc.gov.

Dated at Rockville, Maryland, this 24th day of September 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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