February 6, 2006

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

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2) - ISSUANCE OF AMENDMENTS RE: CONTAINMENT CONVERSION FROM SUBATMOSPHERIC TO ATMOSPHERIC OPERATING CONDITIONS (TAC NOS. MC3394 AND MC3395)

Dear Mr. Lash:

The Commission has issued the enclosed Amendment No. 271 to Facility Operating License No. DPR-66 and Amendment No.153 to Facility Operating License No. NPF-73 for BVPS-1 and 2. These amendments consist of changes to the technical specifications in response to your application dated June 2, 2004, as supplemented February 11, May 12, October 31, and November 14, 2005.

These amendments approve conversion of the BVPS-1 and 2 containments from subatmospheric to atmospheric operating conditions and also approve the Modular Accident Analysis Program - Design-Basis Accident computer code for the BVPS-1 and 2 containment integrity analysis and changes to mass and energy calculation methodologies.

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

Sincerely,

/RA/

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

Docket Nos. 50-334 and 50-412

Enclosures: 1. Amendment No. 271 to DPR-66
2. Amendment No. 153 to NPF-73
3. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

<u>DISTRIBUTION</u>:

PUBLIC	LPLI-1 R/F	RLaufer	TColburn	SLittle	JNakoski
OGC	ACRS	GHill(4)	TBoyce	RLobel	RDennig
RBellamy, RI	DORL DPR	RLobel	VKlein	AHowe	RJenkins

ACCESSION NO. ML060100325 *Safety evaluation dated 12/19/05. No substantive changes made.

OFFICE	LPLI-1/PM	LPLI-1/LA	SPWB/BC	ACVB/BC	ITSB/BC	LPLI-1/BC	OGC
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DATE	1/20/06	1/19/06	12/19/05	12/19/05	2/02/06	2/01/06	1/25/06

OFFICIAL RECORD COPY

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FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 271
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 June 2, 2004, as supplemented February 11, May 12, October 31, and November 14, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 271, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to Mode 4 entry during startup from 1R17 which begins on or about February 10, 2006.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: February 6, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 271

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

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
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-29	3/4 3-29
3/4 3-29a	3/4 3-29a
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 6-5b	3/4 6-5b
3/4 6-6	3/4 6-6
3/4 6-7	------
3/4 6-8	3/4 6-8
3/4 6-9	------
3/4 6-16	3/4 6-16
3/4 6-18	3/4 6-18
3/4 6-19	------
3/4 6-25	3/4 6-25

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 153
License No. NPF-73

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee), dated June 2, 2004, as supplemented February 11, May 12, October 31, and November 14, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-73 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 153, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to Mode 4 entry during startup from 2R12 which begins October 2006.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: February 6, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 153

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

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
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-33	3/4 3-33
3/4 3-34	3/4 3-34
3/4 3-35	3/4 3-35
3/4 6-5a	3/4 6-5a
3/4 6-6	3/4 6-6
3/4 6-7	------
3/4 6-8	3/4 6-8
3/4 6-16	3/4 6-16
3/4 6-17	------
6-26	6-26

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 271 AND 153 TO FACILITY OPERATING

LICENSE NOS. DPR-66 AND NPF-73

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

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

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By application dated June 2, 2004, Agencywide Documents Access and Management System (ADAMS) Accession No. ML041560461, as supplemented February 11, May 12, October 31, and November 14, 2005 (ADAMS Accession Nos. ML050480440, ML051370431, Ml053110142, and ML053220247), the FirstEnergy Nuclear Operating Company (FENOC, the licensee), requested changes to the Technical Specifications (TSs) for BVPS-1 and 2. The supplements dated February 11, May 12, October 31, and November 14, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 20, 2004 (69 FR 43462).

The proposed changes would revise the BVPS-1 and 2 TSs to permit each unit to be operated with an atmospheric containment design. The proposed changes also approves the Modular Accident Analysis Program - Design-Basis Accident (MAAP-DBA) computer code for the BVPS-1 and 2 containment integrity analysis and changes to mass and energy calculation methodologies.

These changes are part of the licensee's Full Potential Program which consists of four elements. These are:

- Containment conversion of both the BVPS-1 and 2 containments from subatmospheric to atmospheric;

C Application of the Westinghouse best-estimate large-break loss-of-coolant accident (BELOCA) methods to BVPS-1 and 2;

C Implementation of an extended power uprate (EPU) which will increase the reactor core power level from 2689 megawatt thermal (MWt) to 2900 MWt for both BVPS-1 and 2; and,

C Installation of replacement steam generators (RSGs) on BVPS-1

Only the first element, the containment conversion, is addressed in this safety evaluation (SE). However, the licensee has performed the supporting analyses at the power level corresponding to the EPU (2900 MWt plus a 0.6 percent uncertainty), and has considered both the current and the RSGs for BVPS-1.

Normal operation with the current design and licensing bases requires the containment air partial pressure be maintained at a vacuum of between 8.9 psia (BVPS-1) or 9.0 psia (BVPS-2) and 10.5 psia (as a function of service water temperature) to limit the peak pressure following a design-basis accident (DBA). This limits the release of fission products following a LOCA inside containment. The containment is accessible for inspection and minor maintenance during operation without altering this pressure provided personnel employ oxygen-enriched breathing equipment.

Revising the containment design-basis revises the response to the design-basis LOCA. Currently, following the design-basis LOCA, the containment pressure must be reduced to subatmospheric pressure within 60 minutes. The revision to the containment design basis will revise this requirement to require that the containment accident pressure must not exceed the containment design pressure or containment liner temperature (45 psig and 280 EF).[1] The containment design temperature and pressure themselves remain unchanged by the containment conversion. The licensee has not revised the containment structural analyses. In addition, the revision to the design and licensing bases for the dose analyses will credit a 50 percent reduction in the containment leakage rate after the first 24 hours, as specified in Regulatory Guide (RG) 1.183.[2]

The BVPS-1 and 2 containment structures are cylindrical, steel-lined, reinforced concrete vessels with flat bases and hemispherical domes which enclose, respectively, the components and major piping comprising the BVPS-1 and 2 reactor coolant system (RCS) pressure boundaries.

As previously stated, both BVPS-1 and 2 are currently licensed to operate at a power level of 2689 MWt. The licensee has separately requested an increase in the licensed rated thermal

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1 BVPS-1: Updated Final Safety Analysis Report Section 5.2.2, "Design Basis and Loading Criteria," BVPS-2: Updated Final Safety Analysis Report Table 6.2-3, "Containment Design Evaluation Parameters."

2 "Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183, Appendix A, Paragraph 3.7, USNRC, July 2000.

power of both units to 2900 MWt[3] and has, therefore, performed the analyses for the containment conversion at the increased power level.

The licensee also requests Nuclear Regulatory Commission (NRC) approval of the analysis methods used to support the proposed containment conversion. These methods are described in Enclosure 2 to the licensee's June 2, 2004, letter, and in a previous letter dated November 24, 2003.[4] The analysis methods consist of the use of NRC-approved methods for calculating the mass and energy release to the containment for the first hour following a design-basis large-break LOCA[5] and for the mass and energy release following a design-basis main steamline break (MSLB) accident.[6] The licensee proposes using the MAAP-DBA computer code to calculate the energy release to the containment 1 hour after the large-break LOCA and for the small- and intermediate-break LOCAs. The licensee proposes using NRC-approved Westinghouse methods to calculate the mass release from a LOCA for times greater than 1 hour.[7]

The MAAP-DBA computer code will also be used to calculate the containment response to the LOCA and the MSLB accident. MAAP-DBA has not been previously approved by the NRC. The licensee's November 24, 2003, letter, provides a description of the containment analysis methods. Section 9 of Enclosure 2 of the licensee's June 2, 2004, letter provides a further description of MAAP-DBA. This SE provides the NRC staff's evaluation of MAAP-DBA for the mass and energy release and containment analyses proposed by the licensee.

The proposed changes to the BVPS-1 and 2 TSs are provided in Attachments A-1 and A-2 to Enclosure 1 of the licensee's June 2, 2004, letter, for BVPS-1 and 2, respectively.

The corresponding changes to the TS Bases are provided in Attachments B-1 and B-2 to Enclosure 1 of the licensee's June 2, 2004, letter for BVPS-1 and 2, respectively. The Bases

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3 FirstEnergy Nuclear Operating Company (FENOC), "Beaver Valley Power Station, Unit No.1 and Unit No. 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, License Amendment Request Nos. 302 and 173, October 24, 2004.

4 FirstEnergy Operating Company (FENOC), "Beaver Valley Power Station, Unit No. 1 and Unit No. 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket 50-412, License No. NPF-73, Containment Conversion Pre-Application Report," FENOC letter to NRC L-03-188, November 24, 2003.

5 WCAP-10325-P-A, (Proprietary), WCAP-10326-A (Non-proprietary), "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," May 1983.

6 WCAP-8822 (Proprietary) and WCAP-8860 (Non-proprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-s1-P-A (Proprietary) and WCAP 8860-S1-A (Non-proprietary), "Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP 8860–S2-A (Non-proprietary), "Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

7 WCAP-10325-P-A, (Proprietary), WCAP-10326-A (Non-proprietary), "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," May 1983.

changes do not require NRC review and approval. They are covered by the licensee's Technical Specification Bases Control Program (see Sections 6.18 of the BVPS-1 and 2 TSs).

The licensee also provided, in Attachments C-1 and C-2 to Enclosure 1 of the licensee's June 2, 2004, letter, for BVPS-1 and 2, respectively, the proposed changes to the Licensing Requirements Manual. These changes also do not require NRC review and approval. The BVPS-1 and 2 Licensing Document Control Program[8] controls the review, approval and implementation of these changes.

The following changes proposed by the licensee require prior NRC review and approval:

1. The allowable values (AVs) of the containment pressure high (Phase A containment isolation), containment pressure high-high (Phase B containment isolation), and the containment pressure intermediate high-high engineered safety features actuation system (ESFAS) setpoints are revised (Items 1.c, 2.c, 3.b.3, and 4.c) in Table 3.3-3 of TS 3.3.2.1, "Engineered Safety Features Actuation System Instrumentation." This change applies to the TSs for both BVPS-1 and 2.

2. The AV for Refueling Water Storage Tank (RWST) Level-Low (BVPS-1, TS Table 3.3-3, Item 1.1.c and BVPS-2, TS Table 3.3-3, Item 1.1.b) are revised.

3. TS Table 3.3-3, Item 1.1.d, "Refueling Water Storage Tank Level - Auto QS [quench spray] Flow Reduction," is deleted from the table along with the associated surveillance requirements (SRs) for this function in Table 4.3-2, "Engineering Safety Feature Actuation System Instrumentation Surveillance Requirements." This change applies only to BVPS-1.

4. The value of P_a is revised in TS 3.6.1.3, "Containment Air Locks," and in SR 4.6.1.3. P_a is also revised in TS 6.1.7, "Containment Leakage Rate Testing Program." The BVPS-1 value of P_a is revised from 40.0 psig to 43.3 psig and the BVPS-2 value is revised from 44.7 psig to 44.9 psig.

5. TS 3.6.1.4, "Internal Pressure," is revised by replacing the air partial pressure requirement with a total air pressure requirement of $12.8 psia and # 14.2 psia. This change eliminates the need for TS Figure 3.6-1 which the licensee proposes to delete. This change is consistent with the proposed revision from a subatmospheric containment to an atmospheric containment and applies to the TSs of both BVPS-1 and 2.

6. TS 3.6.1.5, "Air Temperature," for the containment average air temperature is revised to replace the existing average air temperature requirement based on TS Figure 3.6-1, which is deleted as part of the

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[8] The licensee controls changes made to procedures, the Updated Final Safety Analysis Report (UFSAR), and the License Requirements Manual through Administrative Procedure, ADM-2206, "Licensing Document Update."

change to an atmospheric containment, with an average air temperature range of $ 70 EF and # 105 EF. In addition, SR 4.6.1.5 is revised by relocating the specific temperature sensor location details for determining the containment average air temperature to the TS Bases. This change applies to both BVPS-1 and 2.

7. The quench spray cutback feature for BVPS-1 will be disabled. BVPS-2 does not have a quench spray cutback. The current BVPS-1 quench spray system design includes a restricting orifice in parallel with a motor-operated cutback valve installed downstream of each quench spray pump. This feature extends the delivery time of quench spray which is necessary to ensure subatmospheric pressure following a DBA. With the conversion to an atmospheric containment this feature is not required and is not credited in the accident analyses. Therefore, TS 3.6.2.3, “Chemical Addition System,” is revised by deleting SR 4.6.2.3.d.4 which verifies that one of two operating chemical addition pumps in each subsystem will stop following closure of the BVPS-1 cutback control valve. This change is related to TS change 3. above.

8. TS 3.6.3.1, “Containment Isolation Valves,” is revised to delete SRs 4.6.3.1.b and 4.6.3.1.e which deal with certain containment isolation check valves which ensure containment isolation at subatmospheric conditions. This change applies to both BVPS-1 and 2.

9. The use of the MAAP-DBA computer code for the analysis of the containment response to LOCA and MSLB accidents.

In addition to these TS changes and the revised analysis methods, the licensee’s June 2, 2004, letter, describes several commitments to plant modifications related to the conversion from a subatmospheric to an atmospheric containment which the licensee has concluded, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.59, do not require NRC prior review and approval, but are credited in the safety analyses for the containment conversion. These changes are:

1. Cavitating venturi flow elements will be installed in each auxiliary feedwater (AFW) system injection line in BVPS-1. Such flow elements are currently installed in the BVPS-2 AFW system. The purpose of these flow elements is to limit the mass and energy released into the containment for a postulated MSLB accident at the proposed EPU power level.

   These cavitating venturis are also credited for mitigating breaks outside the containment as part of the analyses supporting the EPU. This aspect of the NRC staff’s review will be addressed as part of the EPU review and is not addressed in this SE.

The licensee states that the design of the cavitating venturis is such that the minimum AFW flow for anticipated transients without scram (ATWS) continues to be met. (Section 7.14 of Enclosure 2 to the licensee's June 2, 2004, letter).

2. Rapid closing feedwater isolation valves will be installed in each of the main feedwater lines in BVPS-1. These valves will limit the mass and energy release into the containment from the postulated MSLB accident at the proposed EPU power level with the containment conversion. BVPS-2 is already equipped with rapid-closing feedwater isolation valves.

3. Drainage ports will be installed in the concrete reactor cavity wall of both BVPS-1 and 2. The drainage port will minimize the water holdup in the reactor cavity which will increase the sump inventory.

The licensee stated that the regulatory commitments contained in the licensee's June 2, 2004, submittal, are to be carried out prior to amendment implementation (see Attachment D of Enclosure 1 of the June 2, 2004, submittal).

2.0 <u>REGULATORY EVALUATION</u>

The current design bases for the BVPS-1 and 2 containments consist of the following:

C Peak pressure associated with a DBA shall be less than the containment design pressure (45 psig).

C The containment shall be depressurized to subatmospheric pressure within 60 minutes following a DBA.

C Once depressurized, the containment shall remain subatmospheric for the duration of the accident.

C Peak containment liner design-basis temperature shall not exceed 280 EF.

C Subcompartment walls shall withstand current design-basis breaks within subcompartments.

C Inadvertent actuation of the containment spray shall not result in a containment pressure less than the minimum containment pressure of 8 psia.

Redefining the BVPS-1 and 2 containment designs as atmospheric containments requires revising the design bases to the following:

C Peak pressure associated with a DBA shall continue to be less than containment design pressure (45 psig).

C Containment pressure is reduced to less than 50 percent of the peak calculated LOCA pressure (P_a) within 24 hours following a DBA.

C Peak containment liner temperature shall continue to not exceed 280 EF.

C Subcompartment walls shall continue to withstand current design-basis breaks within subcompartments.

C Inadvertent actuation of the containment spray shall continue to not result in a containment pressure less than the minimum containment pressure of 8 psia.

The revised design basis is consistent with that for an atmospheric containment and is acceptable.

The general design criteria (GDC) of Appendix A to 10 CFR Part 50 are part of the licensing bases of both BVPS-1 and 2. The GDCs affected by the proposed containment conversion are GDCs 4, 16, 38 and 50.

GDC 4 requires that structures, systems and components be capable of withstanding the environmental conditions associated with postulated accidents, including a LOCA. Dynamic effects associated with certain pipe ruptures may be excluded. The analyses associated with the containment conversion (particularly the assumed increase in reactor power) could affect the envelope of conditions under which safety-related equipment was qualified and the effects of pipe ruptures within containment on containment structures.

GDC 16 requires the containment to provide an "essentially leaktight barrier." The change from a subatmospheric containment to an atmospheric containment will not change this requirement. The licensee has not proposed changing the TS requirement for the allowable containment leakage rate, L_a, although the duration of the leakage is no longer limited to 1 hour and the containment leakage rate may, under the new design and licensing basis, be reduced to 50 percent of L_a after 24 hours.

GDC 38 requires a system capable of rapidly reducing the containment temperature and pressure following a LOCA and maintaining them at acceptably low levels assuming the worst single failure. This license amendment request proposes changing the roles of the quench and recirculation sprays. With the change to an atmospheric containment, the containment pressure will not be reduced to subatmospheric within an hour. The containment pressure must be reduced by 50 percent within 24 hours to comply with the proposed licensing basis.

GDC 50 requires that the containment structure and heat removal systems must accommodate, with sufficient margin, the pressure and temperature resulting from a LOCA without exceeding the containment leakage rate. The proposed change in the containment design basis results in an increased calculated containment pressure and temperature. However, these values remain less than the containment design temperature and pressure and, as stated above, L_a is not changed.

The current licensing basis for emergency pump net positive suction head (NPSH) for BVPS-1 credits containment accident pressure in determining the available NPSH for the recirculation spray and low-head safety injection (SI) pumps. The licensee proposes to retain credit for containment accident pressure for these BVPS-1 pumps, and to utilize the MAAP-DBA computer code to determine the containment accident pressure, sump water level and sump water temperature, which are inputs to the NPSH calculations. For BVPS-2, only the recirculation spray pumps take suction from the recirculation sump. After the injection phase, two of the four recirculation spray pumps inject into the reactor vessel cold legs. The NPSH calculations for BVPS-2 currently do not credit containment accident pressure in determining available NPSH for the recirculation spray pumps. The current licensing basis for BVPS-2 assumes that the containment pressure is equal to the vapor pressure at the calculated recirculation sump water temperature. This assumption is consistent with the guidance of Standard Review Plan (SRP), Section 6.2.2.[9] This licensing basis will continue for BVPS-2, although MAAP-DBA will be used to determine the sump water level and sump water temperature.

The licensing basis radiological dose analyses for BVPS-1 and 2 are based partially on the accident source term requirements of 10 CFR 50.67 and RG 1.183 and partially on Technical Information Document 14844 and the SRP. This is not changed by this proposed license amendment request. The status of the BVPS-1 and 2 licensing basis with respect to radiological dose is summarized in Sections 1.4.2 and 1.5.2 of Enclosure 2 to the licensee's June 2, 2004, letter.

In order to ensure that TS and licensing basis limits are not exceeded, the analyses supporting operation must include the uncertainty in the parameters specified in the TSs and in parameters not in the TSs which can significantly affect TS and design-basis limits. The licensee has addressed this subject in describing the safety analyses in Enclosure 2 to the June 2, 2004, letter, and in the February 11, and May 12, 2005, letters. (See especially the table provided in response to the NRC staff's Request for Additional Information (RAI) Question No. 17, in the licensee's February 11, 2005, letter).

3.0 TECHNICAL EVALUATION

Enclosure 2 of the licensee's June 2, 2004, letter, provides re-analyses of the DBAs applicable to BVPS-1 and 2 to demonstrate that the applicable safety and regulatory criteria are still satisfied following conversion from subatmospheric to atmospheric containments at EPU conditions. The re-analyses also consider the replacement of the BVPS-1 Westinghouse Model 51 steam generators (SGs) with Westinghouse Model 54F SGs. Thus, the Enclosure 2 calculations include three aspects of the licensee's Full Potential Program even though the June 2, 2004, letter, only requests approval of the conversion of the BVPS-1 and 2 containments from subatmospheric to atmospheric operating conditions.

The scope of the analyses covered by this review, as specified by the licensee in Enclosure 2 to the June 2, 2004, letter, includes the peak containment pressure, the temperature of the containment atmosphere, the containment liner temperature, long-term sump water

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[9] Standard Review Plan Section 6.2.2, "Containment Heat Removal Systems," Revision 4, USNRC, October 1985.

temperature, the available NPSH for the emergency core cooling system (ECCS) and containment spray pumps, and the maximum service water outlet temperature for the containment heat removal heat exchanger. The minimum containment pressure due to inadvertent containment spray actuation was also re-analyzed as described in Enclosure 2 of the licensee's June 2, 2004, letter, and the licensee's May 12, 2005, letter.

The licensee proposes the following changes to the containment analysis methods:

- Currently, the mass and energy release licensing basis for BVPS-1 is the Stone and Webster LOCTIC[10] code for the LOCA blowdown phase and Westinghouse methods described in WCAP-10325-P-A[11] for the reflood and post-reflood periods. WCAP-10325-P-A has previously been approved by the NRC. The licensee proposes changing the BVPS-1 mass and energy release calculation method from the LOCTIC computer code to the Westinghouse methods of WCAP-10325-P-A to model the first hour of the LOCA. After the first hour, the MAAP-DBA computer code will be used to calculate the energy release for the remainder of the accident. The mass release will still be calculated with WCAP-10325-P-A. MAAP-DBA is described in Section 9.0 of Enclosure 2 to the licensee's June 2, 2004, letter, and in the licensee's November 24, 2003, letter to the NRC.[12]

- Currently, BVPS-2 mass and energy release calculations are performed using the methods of WCAP-10325-P-A. For the containment conversion, the mass and energy release calculations will be performed in the same way as those for BVPS-1 described above.

- The small-break LOCA mass and energy releases will be calculated with the MAAP-DBA computer code.

- The containment response to a small-break LOCA will be calculated with the MAAP-DBA code.

- The LOCA and MSLB containment integrity and available NPSH calculation methods will be changed from LOCTIC to MAAP-DBA.

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10 "LOCTIC - A Computer Code to Determine the Pressure and Temperature Response of Dry Containment to a Loss-of-Coolant Accident," Stone and Webster Engineering Corporation, September 1971 (Proprietary).

11 WCAP-10325-P-A, (Proprietary), WCAP-10326-A (Non-proprietary), "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," May 1983.

12 FirstEnergy Operating Company (FENOC), "Beaver Valley Power Station, Unit No. 1 and Unit No. 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket 50-412, License No. NPF-73, Containment Conversion Pre-Application Report," FENOC letter to NRC L-03-188, November 24, 2003.

C The MSLB accident mass and energy release calculation methods will be changed to the Westinghouse methods of WCAP-8822.[13] MAAP-DBA will be used for the MSLB containment integrity calculations.

The following table, based on Table ES in the licensee's November 24, 2003, letter, summarizes the methods the licensee proposes to use.

Application of Analytic Methods to Containment Conversion		
Parameter	Mass/Energy	Containment
LOCA: Pressure Gas Temperature Containment liner temperature	Less than 1 hour: NRC-Approved WCAP-10325-P-A Greater than 1 hour: Mass from WCAP-10325-P-A. Energy from MAAP-DBA	MAAP-DBA: Single Node Tagami/Uchida Heat Transfer No entrainment 10% airborne water
MSLB: Pressure Gas temperature Containment liner temperature	NRC-Approved WCAP-8822-P-A (w/Supplements 1 & 2)	MAAP-DBA Single Node Uchida heat transfer No entrainment 8% revaporization
LOCA: NPSH: sump water inventory and temperature Maximum service water outlet temperature	NRC-Approved WCAP-8264-P-A, Revision 1	Multiple node MAAP-DBA
SBLOCA: Containment Pressure NPSH: Sump Water inventory and temperature Maximum service water outlet temperature	MAAP-DBA	Multi node MAAP-DBA

The licensee's calculations of the DBAs include the proposed changes to the BVPS-1 and 2 TSs and the proposed plant modifications discussed in Section 1.0 of this SE.

The licensee states (Enclosure 2 of the licensee's June 2, 2004, letter, Section 4.1.1) that the design-basis containment analyses are implemented consistent with the SRP. In general, the NRC staff agrees. However, the licensee has also used more current models and assumptions than those of the SRP. These were reviewed and are evaluated in this SE.

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13 WCAP-8822 (Proprietary) and WCAP-8860 (Non-proprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP 8860-S1-A (Non-proprietary), "Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP 8860–S2-A (Non-proprietary), "Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

The technical evaluation of the proposed changes to the BVPS-1 and 2 licensing bases are discussed in the following subsections.

3.1 Nuclear Steam Supply System (NSSS) Parameters
3.2 Large-Break LOCA
3.2.1 Large-Break LOCA Mass and Energy Release
3.2.2 Large-Break LOCA Containment Response
3.3 Small- and Intermediate-Break LOCAs
3.3.1 Small- and Intermediate-Break LOCAs Mass and Energy Release
3.3.2 Small- and Intermediate-Break LOCAs Containment Response
3.4 Subcompartment Analysis
3.5 MSLB Accident
3.5.1 MSLB Mass and Energy Release
3.5.2 MSLB Accident Containment Analysis
3.6 NPSH of Recirculation Spray and Low-Head SI Pumps
3.7 Inadvertent Containment Spray Actuation
3.8 Equipment Qualification Envelope
3.9 Combustible Gas Monitoring and Control
3.10 MAAP-DBA Evaluation
3.10.1 Comparison with GOTHIC (Generation of Thermal Hydraulic Information for Containments) 6.0
3.10.2 Separate Effects Tests
3.10.3 Integral Tests
3.10.4 Conclusion of MAAP-DBA Validation
3.11 TS Changes
3.12 Containment Isolation
3.12.1 Containment Leakage Rate Testing
3.12.2 Containment Purge Supply and Exhuast and Vacuum Ejector Containment Penetrations
3.12.3 Hydrogen Recombiner Discharge Piping Containment Penetrations
3.12.4 Recirculation Spray Heat Exchanger River Water Monitor Piping Containment Isolation
3.12.5 GL 96-06
3.13 Radiological Assessment
3.14 Station Blackout
3.15 Other Considerations
3.15.1 UFSAR Accident Analyses
3.15.1.1 LOCA Transients
3.15.1.1.1 Large-Break LOCA
3.15.1.1.2 10 CFR 50.46 Large-Break Reporting Requirements
3.15.1.1.3 Small-Break LOCA
3.15.1.1.4 10 CFR 50.46 Small-Break Reporting Requirementws
3.15.1.1.5 SG Tube Rupture
3.15.1.2 Non-LOCA Transients
3.15.1.2.1 MSLB
3.15.1.2.2 Feedline Break
3.15.1.2.3 ATWS
3.15.1.2.4 Natural Circulation

3.15.1.3 Initial Condition Uncertainties
3.15.1.4 ECCS
3.15.1.5 UFSAR Accident Analyses Conclusions
3.15.2 NSSS System Components and Nuclear Fuel
3.15.3 Safety-Related Valves
3.15.4 Supplementary Leak Collection and Release System (SLCRS)
3.15.5 Quench Spray System
3.15.6 Recirculation Spray System
3.15.7 Chemical Addition System
3.15.8 Piping Systems
3.15.9 River Water System
3.15.10 Service Water System
3.15.11 Main Feedwater System
3.15.12 AFW System
3.15.13 Instrumentation

3.1 Nuclear Steam Supply System (NSSS) Parameters

The calculations for the containment conversion and EPU use input parameters applicable to both. The licensee terms these Performance Capability Working Group parameters. These parameters for BVPS-1 also include parameters for the Model 54F RSGs. These parameters are listed in Tables 2.1-1, 2.1-2, and 2.1-3 of Enclosure 2 to the licensee's June 2, 2004, letter. The licensee states that these parameters form the basis for the subsequent analyses and evaluations of the containment conversion. In order to bound expected operating conditions, these parameters cover a range of primary and secondary side conditions. The licensee states that these parameters for the containment conversion project bound the current operating conditions as well as the EPU conditions. In a February 11, 2005, response to an NRC staff RAI, the licensee stated that these values are restricted to certain fuel types by core pressure drop, bypass flow, power distribution limits, etc. Therefore, a change in core configuration could result in a change to the containment licensing basis.

The acceptability of these NSSS parameters depends on the acceptability of the mass and energy release and containment analyses which use them and any TS restrictions. These are discussed in the remainder of this SE.

3.2 Large-Break LOCA

3.2.1 Large-Break LOCA Mass and Energy Release

The uncontrolled discharge of primary or secondary coolant into the containment is termed a mass and energy release.

The LOCA mass and energy release analysis for the containment conversion assumes the EPU core power, including calorimetric error, of 2917.4 MWt (2900 MWt X 1.006).

The licensee states that this analysis bounds both the Model 51 and the planned (Model 54F) RSGs for BVPS-1 and the Model 51M SGs for BVPS-2.

The current BVPS-1 LOCA licensing basis mass and energy release calculation uses the LOCTIC computer code.[14] Westinghouse methods are used for the reflood and post-reflood response. The current BVPS-2 licensing basis LOCA mass and energy release analyses use Westinghouse March 1979 methods.[15]

The licensee proposes to use the NRC-approved Westinghouse March 1979 methods (WCAP 10325-P-A) in the containment conversion project for the mass and energy input to the calculation of the first hour of the containment response to the large-break LOCA. Sections 3.1.2 to 3.1.5.7 of Enclosure 2 to the licensee’s June 2, 2004, letter, provide an overall description of this model. Following the first hour, the coolant enthalpy for the remainder of the large-break LOCA is calculated with the MAAP-DBA code while the mass release continues to be calculated with the methods of WCAP 10325-P-A (the licensee states that at this time in the accident, the mass flow discharged from the RCS has reached equilibrium with the injection rate). This is described in response to an NRC RAI in the licensee’s February 11, 2005, letter.[16] The licensee states that calculating the enthalpy with MAAP-DBA produces a more accurate release enthalpy since the Westinghouse methods use a fixed sump temperature that does not incorporate long-term sump cooling. In order to study this approach, the NRC staff performed independent calculations of the mass and energy release after 1 hour using the RELAP5 computer code. The results of the NRC staff’s calculations are provided in the Appendix 1 of this SE. The calculations show close agreement between the NRC staff’s calculations and the licensee’s for mass flow rate, energy release and the integrated energy discharged into the containment as a function of time. The NRC staff therefore, finds the licensee’s approach of calculating the mas flow with the methods of WCAP 10325-P-A and the energy release with MAAP-DBA to be acceptable.

The LOCA mass and energy release calculation contains conservative assumptions. These are listed in Section 3.1.2 of Enclosure 2 to the licensee’s June 2, 2004, letter, and also discussed in the licensee’s February 11, 2005, letter. These assumptions overestimate the steady state energy available to be released, the mass available to be released and the mass flow rate into the containment. One of these conservative assumptions, for example, is the RWST water temperature. The licensee assumed an RWST temperature of 65 EF. This is conservative relative to the upper bound TS limit for both units which is 55 EF for BVPS-1 and 50 EF for BVPS-2. The licensee is not proposing to change the RWST temperature TS.

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14 “LOCTIC-A Computer Code to Determine the Pressure and temperature response of Dry Containment to a Loss-of-Coolant Accident,” Stone and Webster Engineering Corporation, September 1971 (Proprietary).

15 WCAP-10325-P-A, (Proprietary), WCAP-10326-A (Non-proprietary), “Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version,” May 1983.

16 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, to USNRC, Response to Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190, February 11, 2005.

The licensee also stated that all parameters included in the analyses are chosen consistent with accepted analysis methodology.[17]

The LOCA mass and energy release calculations assume a loss of all offsite power and a single failure. The licensee assessed two potential worst-case single failures. The first case assumes a loss of one emergency diesel generator resulting, along with the assumed loss of offsite power, in the loss of one train of safeguards equipment. This minimizes the SI flow rate and cooling to the core. The other case assumes failure of a train of containment spray in the containment analysis. These single failures are typically assumed for Westinghouse-designed reactors.

In addition, for BVPS-2, a third case is considered. This is the failure of one train of the service water supply to the recirculation spray heat exchangers. There are two recirculation spray heat exchangers per train. Each pair of heat exchangers in one train is cooled by one train of service water. A portion of the recirculation spray system provides low-head SI flow in the recirculation mode. The two recirculation spray pumps inject sump water into the low-head SI header to supply reactor vessel inventory during the recirculation phase of a LOCA. This recirculation spray pump flow also supplies the high-head SI pumps. In order to establish the maximum temperature of the water pumped into the low-head SI piping to ensure that thermal stresses in this piping remain within allowable values, a single failure of the service water supply to the recirculation spray heat exchangers is assumed. Since BVPS-1 does not have this ECCS design, this failure is only applicable to BVPS-2.

The licensee has acceptably considered all potential sources of mass and energy in accordance with the guidance of SRP Section 6.2.1.3.[18] These are listed in Enclosure 2 to the licensee's June 2, 2004, letter, in Tables 3.1-3, 3.1-14 and 3.1-15 for BVPS-1 and Tables 3.1-39, 3.1-40, 3.1-41 and 3.1-48 for BVPS-2.

The licensee assumed that no zirconium-water reaction would occur and therefore there is no contribution to the total energy released to the containment from this energy source. The licensee states that[19]

> The energy release from the Zr-water [zirconium-water] reaction was not considered as part of the WCAP-8264-P-A, Rev.1 or the WCAP-10325-P-A [mass and energy release] methodology. Based on the manner in which the energy in the fuel is conservatively modeled to be released to the vessel fluid, the fuel cladding temperature does not increase to the point where the Zr-water

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17 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, to USNRC, Response to Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190, Response to RAI 13, February 11, 2005.

18 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," USNRC July 1981.

19 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, to USNRC, Response to Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190, Response to RAI 10, February 11, 2005.

reaction is significant. This is in contrast to the 10 CFR 50.46 analyses, which are biased to calculate high fuel rod cladding temperatures.

There are no acceptance criteria for the predicted mass and energy release. The mass and energy releases are part of the calculation of the containment response to the postulated DBA.

The NRC staff did perform an audit calculation of the mass and energy release for the blowdown period, that is, the time from the pipe break to the emptying of the reactor vessel. The results, calculated using the RELAP5 computer code were in close agreement with those of the licensee using Westinghouse NRC-approved methods. A description of the calculation and the results are provided in Appendix 2 of this SE.

3.2.2 Large-Break LOCA Containment Response

The licensee calculates the containment response to the large-break LOCA with the MAAP-DBA computer code using the mass and energy release input from the approved Westinghouse methods discussed earlier. Section 3.10 of this SE provides the NRC staff's evaluation of MAAP-DBA.

The licensee uses two different containment models for the large-break LOCA. A single node model is used for calculating the peak containment pressure, containment liner temperature and containment atmosphere temperature for environmental qualification. A multiple node model is used for calculations which track the water inventory in various parts of the containment. These calculations are the large-break LOCA NPSH, small-break LOCA NPSH and sump water temperature models. The evaluation of the small-break LOCA containment calculations is provided in Section 3.3 of this SE and the evaluation of the NPSH calculations is provided in Section 3.6. The sump water temperature is also input to the calculation of recirculation spray piping temperatures and recirculation spray heat exchanger cooling water outlet temperature.

The licensee models heat sinks in containment as distributed (single- or double-sided) or as lumped. Thermal resistances due to paint, gaps, etc. are modeled. This is modeled as a standard heat conduction problem and is acceptable.

The single node containment model is consistent with the use of the Tagami[20] and Uchida[21] heat transfer correlations which the licensee uses to calculate the transfer of energy from the containment atmosphere to structures in the containment and the containment walls. The NRC staff considers these heat transfer correlations suitably conservative. They are commonly used in NRC-approved containment licensing calculations.

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20 Takashi Tagami, "Interim Report on Safety Assessment and Facilities Establishment Project in Japan for the period ending June 1965" (No. 1) Unpublished Work, 1965.

21 Uchida, H., Oyama, A., Togo, Y., Evaluation of Post Incident Cooling Systems in Light-Water Power Reactors, Proceedings of the Third International Conference on the Peaceful uses of Atomic Energy held in Geneve, Switzerland, August 31 to September 9, 1964, Volume 13, New York: United Nations, 1965, (A/CONF.28/P/436) May 1964 pp. 93-104.

The multiple node model uses natural convection heat transfer models[22] for heat transfer to containment walls and internal structures. These models are described in Section 2.1.3 of the attachment to the licensee's November 24, 2003, letter. Limiting the model to natural convection (as opposed to inclusion of turbulence) is conservative since the agitated motion of the containment atmosphere due to break flow and spray-induced motion would be expected to produce higher heat transfer rates.

The containment model accounts for the presence of water droplets produced by the LOCA blowdown in the containment atmosphere. These droplets have only a small effect on the containment temperature for the LOCAs but have a significant effect on the temperature of the containment atmosphere for the MSLB. This aspect of the modeling is discussed further in Section 3.10 of this SE.

The licensee models the containment spray as consisting of 1000 micron drops for both the quench spray and the recirculation spray. For the same flow rate, a smaller droplet will remove more heat from the containment atmosphere. Therefore, a larger droplet size is conservative for LOCA analysis. The licensee states that the 1000-micron droplet is consistent with the spray nozzle manufacturer's data. The BVPS-1 UFSAR states that the mean equivalent diameter of the recirculation spray droplets is less than 1000 microns (page 6.4-7) and that a droplet diameter of 2000 microns would still produce a spray effectiveness of greater than 99.9 percent (page 14.3-35). The BVPS-1 UFSAR also states that the spray nozzles are of different sizes to give the optimum combination of small spray droplets for maximum heat transfer and large spray droplets for better coverage toward the center and sides of the containment (UFSAR, page 6.4-10). Thus explaining the need for using a mean diameter for the spray droplets in the analyses. The BVPS-2 UFSAR, Figure 6.4-125, provides the frequency distribution of spray droplet size for BVPS-2. The median droplet size is 230 microns. The most frequent droplet size is approximately 100 microns. The frequency of the 1000-micron droplet is an order of magnitude less than the 100-micron droplet. Thus, the choice of the 1000-micron spray droplet diameter is conservative.

The distribution of the mass and energy of the break flow in the containment atmosphere is an important consideration in determining the peak pressure and the maximum sump temperature and is discussed in Section 3.10 of this SE.

The licensee models heat transfer from water on the containment floor to the containment atmosphere.[23] The licensee has shown that this has little effect on the maximum pressure and temperature. This agrees with the NRC staff's studies.[24]

There are three possible break locations to consider for the large-break LOCA containment analyses. These are the double-ended break in the hot-leg, the double-ended break in the

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22 R. B. Bird, W. E. Stewart, and E. N. Lightfoot, Transport Phenomena, Wiley, New York, 1960.

23 Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, Response to RAI 7, May 12, 2004.

24 Tills, Jack; Notafrancesco, Allen; and Murata, Ken, CONTAIN Code Qualification Report/User Guide for Auditing Design-Basis PWR Calculations, SMSAB-02-03, August 2002, pages 2-5 (ADAMS Accession Number ML022490371).

pump suction pipe and the double-ended break in the cold-leg. The licensee concludes, based on the response of the RCS to the postulated break and the response of the ECCS that the cold-leg break will be bounded by one of the other two breaks and does not need to be considered. This is a typical result for Westinghouse pressurized-water reactors (PWRs) and is acceptable.

The licensee provided a detailed listing of the conservative assumptions made for the important input and modeling parameters in Table 4-3, "Containment Parameter Limits for Design Basis Analysis," in the June 2, 2004, submittal, and discussed the conservatism in the LOCA containment analysis in response to an NRC staff RAI.[25] The NRC staff finds the conservatism in the LOCA containment analysis to be acceptable since it is consistent with approved Westinghouse methods and addresses the significant uncertainties in these analyses.

The following tables, reproduced from Tables 4-16 and 4-17 of Enclosure 2 of the licensee's June 2, 2004, letter, provide the results of the licensee's LOCA containment calculations.

MAAP-DBA Containment Peak Pressure and Temperature LOCA Results BVPS-1				
Description	Power Level (%)	Single failure	Peak Pressure (psig)*	Peak Temperature (EF)
LOCA: DEPS Min SI	100.6	DG+	42.0	266.0
LOCA: DEPS Max SI	100.6	CIB++	42.0	266.0
LOCA: DEHL	100.6	None+++	43.3	267.8

\+ DG: Failure of one train of emergency diesel power which results in the loss of one train of SI and one train each of quench spray and recirculation spray

++ CIB: Failure of one train of quench spray and recirculation spray

+++ No single-failure is assumed since the peak conditions occur prior to operation of the SI and containment sprays

* All pressures referenced to 14.3 psi.

25 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, to USNRC, Response to Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190, Response to RAI 13, February 11, 2005.

MAAP-DBA Containment Peak Pressure and Temperature LOCA Results BVPS-2				
Description	Power Level (%)	Single-failure	Peak Pressure (psig)*	Peak Temperature (EF)
LOCA: DEPS Min SI	100.6	DG	42.4	266.7
LOCA: DEPS Max SI	100.6	CIB	42.4	266.6
LOCA: DEHL	100.6	None	44.9	270.1

* All pressures referenced to 14.3 psi.

The maximum pressure and temperature increases are a result of the double-ended hot-leg break. The peak pressures are less than the containment design pressure and the peak temperatures are less than the 280 EF acceptance criterion.

The peak calculated LOCA pressures shown in the tables, 43.3 psig for BVPS-1 and 44.9 psig for BVPS-2 are the respective values of P_a. The licensee proposes revising the TSs of both units to include these values.

The licensee states in Section 4.7.10 of Enclosure 2 to the June 2, 2004, letter that the containment pressure is reduced to below one-half of the peak pressure within 24 hours.

3.3 Small- and Intermediate-Break LOCAs

The licensee analyzed both cold-leg and hot-leg breaks from 1 to 12 inches in diameter. These breaks were analyzed to determine the response of the containment depressurization system and the impact on long-term cooling as well as the NPSH of the recirculation spray pumps.

3.3.1 Small- and Intermediate-Break LOCA Mass and Energy Release

The licensee uses the MAAP-DBA computer code to calculate the small- and intermediate-break LOCA mass and energy releases. The MAAP-DBA code is evaluated in Section 3.10 of this SE.

The licensee benchmarked MAAP-DBA against two sets of mass and energy release histories generated using the Westinghouse NOTRUMP computer code.[26] The first case compared the MAAP-DBA code with NOTRUMP for BVPS-1 and 2 core response calculations for a range of cold-leg break sizes from 2 to 6 inches. This case used the decay heat model specified by 10 CFR Part 50, Appendix K. The second case compared NOTRUMP and MAAP-DBA for a

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[26] Rupprecht, S.D., et al., “Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code,” WCAP-11145-P-A (Proprietary) 1986.

2-inch cold-leg break using the American National Standards Institute (ANSI) 1979 decay heat standard with a 2σ uncertainty. The agreement with both cases was generally good in both magnitude and trend.

The NRC staff questioned whether it is valid to compare MAAP-DBA with NOTRUMP since neither of these codes has the capability to model multi-dimensional effects. These effects could be important for the larger small breaks. The licensee responded that these effects may impact the mass and energy release for cold-leg break locations but would not have a significant impact on hot-leg breaks of any size.[27] For BVPS-1 and 2, hot-leg breaks were determined to be limiting for all parameters of interest for small and intermediate size breaks except for the minimum sump inventory. The licensee further stated that for the minimum sump inventory, neglecting these effects is conservative since neglecting multi-dimensional effects underestimates the break flow, and hence the water accumulation in containment. The NRC staff concurs with the licensee's response to the NRC staff's question.

The limiting NPSH is obtained for medium- and small-break LOCAs. This is discussed further in Section 3.6 of this SE.

3.3.2 Small- and Intermediate-Break LOCA Containment Response

The licensee states that:

> the results of medium and small-break LOCAs for the balance of containment response attributes [other than NPSH] were all bounded by the LBLOCA [large-break LOCA].

This is the expected result.

3.4 Subcompartment Analysis

The SRP[28] defines a subcompartment as "any fully or partially enclosed volume within the primary containment that houses high energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated piping rupture within the volume." Examples of subcompartments in the BVPS-1 and 2 containments are the SG and pressurizer cubicles. The purpose of subcompartment analyses is to demonstrate that all closed or restricted compartments subjected to these localized pressure buildups are designed so that the resulting pressure differentials between the adjacent compartments are limited to ensure structural capability by not exceeding design loadings.

The licensee states that the effect of EPU on subcompartment analysis is due to the reduction in the RCS temperature. This results in a higher mass flow rate into the subcompartment and consequently a higher rate of pressurization of the subcompartment. The licensee states that

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27 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, to USNRC, Response to Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190, Response to RAI 20(f), February 11, 2005.

28 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Revision 1, Section 6.2.1.2, "Subcompartment Analysis," Revision 2 July 1981.

since the peak pressure is reached within the subcompartment in less than 3 seconds, the effect of power and SG design are not significant. The NRC staff agrees.

The licensee credits leak-before-break considerations for limiting the size of the pipes that must be considered for subcompartment analysis. The licensee states that the reduced size of pipe rupture more than offsets the effect of increased mass flow rate for both units. The licensee provided the results of a BVPS-2 steam generator subcompartment analysis of a split break in an SG inlet elbow to demonstrate this.[29] The current RCS conditions used (as stated in the BVPS-2 UFSAR) are 2,250 psia and 610.8 EF. The break is assumed to be a 707 in^2 split break. The EPU RCS conditions are: 2,280 psia (includes 30-psi uncertainty) and 603.9 EF. The maximum break size is reduced by leak-before-break to a 6-inch diameter. The EPU analysis also conservatively assumes no depressurization of the reactor vessel. The break enthalpy was maximized by assuming 617 EF and 2220 psia (including a 30-psi uncertainty). The calculation demonstrates that the reduced size of the pipe rupture more than offsets the effect of increased mass flow rate for both units.

The NRC staff finds the licensee's subcompartment analyses acceptable for EPU conditions. Since the flow is critical, the results are independent of the containment pressure.
The licensee has not provided a discussion of the use of MAAP-DBA for subcompartment calculations and therefore, the NRC staff approval of MAAP-DBA for BVPS-1 and 2 does not include the use of MAAP-DBA for subcompartment analysis.

3.5 <u>MSLB Accident</u>

3.5.1 MSLB Accident Mass and Energy Release

The rupture of a main steamline inside containment is also a DBA (other than the LOCA) capable of producing high pressure and temperatures within the containment.

The consequences of a MSLB depend on the plant configuration, plant initial operating conditions, the size of the break and the power level as well as the assumed worst single failure. Section 3.2 of the Enclosure 2 to the licensee's June 2, 2004, letter, describes the methods proposed for calculating the mass and energy releases for the MSLB accident. The proposed method considers a spectrum of power levels, break sizes and locations and assumed single failures. The analyses are performed with analysis methods previously approved by the NRC for this type of analysis. The assumptions made by the licensee overestimate the mass and energy released to the containment.

The licensee has chosen break sizes to bound the possible cases. These are:

1. For BVPS-1 with the current (Model 51) SGs: the double-ended rupture upstream of the inline flow restriction in one steam line. This break produces the highest flow rate from the faulted SG for this design. (The Model 51 SG does not have a flow restriction at the SG nozzle.)

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29 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, to USNRC, Response to Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190, RAI 6, February 11, 2005.

2. For BVPS-1 with the replacement (Model 54F) SGs and for BVPS-2 (with Model 51M SGs), both of which have flow restrictions integral with the outlet nozzles of the SGs: A full double-ended rupture of a main steam line. This break produces the highest flow rate from the faulted SG for this design.

3. A small split rupture that does not generate a steamline isolation signal nor result in water entrainment in the break flow. The licensee states that this postulated break does not cause a direct steamline isolation signal. The reactor is tripped and SI is initiated from a containment high pressure signal. The SI signal results in steam line isolation.

WCAP-8822[30] describes the Westinghouse methods used to calculate the mass and energy release from an MSLB. It discusses four break sizes. The licensee has only analyzed two (the double-ended rupture and the small split break). The licensee's February 11, 2005, letter, states that this is sufficient because the two break sizes not analyzed result in water entrainment in the break flow which decreases the enthalpy of the break flow and, therefore, decreases the containment pressure so that these cases are not limiting. The actual break cases for BVPS-1 and 2 are given in Sections 3.2.3.1 and 3.2.3.2 of Enclosure 2 to the licensee's June 2, 2004, letter.

Note that for BVPS-1 in its current configuration, with inline flow restrictions in the steamlines, entrainment is assumed for the large double-ended ruptures. Section 3.2.2.7 of Enclosure 2 to the licensee's June 2, 2004, letter, states that this is consistent with the NRC-approved Westinghouse methods for calculating mass and energy release following an MSLB. The NRC staff agrees.

The MSLB accident mass and energy release calculations for both units include consideration of the worst single-failure, as discussed in Section 3.2.2.2 of Enclosure 2 to the licensee's June 2, 2004, letter. The licensee considered possible failure-to-close of a main steam check valve or a feedwater isolation valve for BVPS-1 and failure-to-close of a main steam isolation valve or a main feedwater isolation valve for BVPS-2. The closure of other valves in the steam or feedwater lines was credited together with their closure times. This is consistent with the SRP position that credit for nonsafety-related equipment is permissible when assuming the single-failure of safety-related equipment.[31] The increase in mass released due to the increase in pipe length if the analysis depended on the closure of other valves at different positions along

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30 WCAP-8822 (Proprietary) and WCAP-8860 (Non-proprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP 8860-S1-A (Non-proprietary), "Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP- 8860–S2-A (Non-proprietary), "Supplement 2 - Impact of Steam Superheat in Mass/Energy. Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986. WCAP-8821-P-A (Proprietary) and WCAP-8859-A (Non-Proprietary) "TRANFLO Steam Generator Code Description," September 1976; approved Version June 2001.

31 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Revision 1, Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," July 1981.

the steamline is accounted for in the licensee's analyses. Feedwater addition continues until terminated by operator or automatic action or closure of these other valves. This is in agreement with NRC Bulletin 80-04[32] and the SRP[33] and is acceptable. The licensee states that the volume of fluid in the feedwater lines that can flash upon depressurization of the faulted SG is maximized, thus further increasing the containment pressure.

The licensee conservatively neglects purging of the AFW system piping prior to injection of cold AFW since this results in colder, more dense, AFW being added to the faulted SG which, in turn, results in increased heat transfer to the secondary coolant and more mass released to the containment. The licensee maximizes flow to the faulted SG and minimizes flow to the intact SG (see the licensee's June 2, 2004, letter, Enclosure 2, Section 3.2.2.4). The licensee assumes that AFW flow to the faulted SG can be terminated by the operator within 30 minutes after accident initiation. This time is consistent with the BVPS-1 and 2 licensing basis.

The licensee assumes the maximum initial SG mass in the faulted-loop SG for all cases, even those at power levels for which the programmed SG level would be different. This increases the mass released to the containment.

The break flow is calculated with the Moody critical flow model.[34] Piping resistance (which would decrease the flow rate into containment) is not considered. These assumptions are consistent with the guidance of SRP Section 6.2.1.4.[35]

The licensee conservatively assumes unrestricted flow through the valves as they are stroking closed to isolate the main steam lines.

The licensee has changed the decay heat standard used for these calculations from the 1971 ANS decay heat standard[36] to the 1979 decay heat standard.[37] The 1979 ANS decay heat standard has previously been approved by the NRC for non-LOCA accident analyses including the analysis of MSLB accidents.

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[32] Analysis of a PWR Main Steam Line Break With Continued Feedwater Addition, NRC Bulletin 80-04, February 8, 1980.

[33] Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Revision 1, Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," July 1981.

[34] F.J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," J. of Heat Transfer, Trans. Am. Soc. of Mechanical Engineers,, Vol. 87, No.1, February 1965.

[35] Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Revision 1, Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," July 1981.

[36] American Nuclear Society Standard 5, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," ANS Standards Committee, October 1971.

[37] ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

In the NRC staff's SE report on the Westinghouse topical report WCAP-8822[38] which describes Westinghouse methods for analyzing the MSLB accident, the NRC staff noted that the effect of heat transfer to steam from the uncovered portion of the SG tube bundle (which superheats the steam) may be significant especially for the ice condenser containment. Supplement 2 to WCAP-8822 demonstrated that this was a minor effect for both the subatmospheric and the large dry containments. The NRC SE report on WCAP-8822 agreed with this finding. Therefore, the effect of superheated steam is not a consideration in the conversion of BVPS-1 and 2 to containments atmospheric containments.

The current MSLB accident mass and energy release analyses used the Westinghouse MARVEL code. For the containment conversion analyses at EPU conditions, the licensee has used the Westinghouse LOFTRAN computer code. The use of the LOFTRAN code for MSLB accident mass and energy release analyses has been previously approved by the NRC.[39]

Table 3.2-4 of Enclosure 2 to the licensee's June 2, 2004, letter, lists the protection system actuation signals and safety system setpoints for the containment conversion and EPU conditions. Since the MSLB accident acceptance criteria are satisfied with the revised setpoint values, as discussed below, they are acceptable.

The licensee assumes failure of one train of SI which results in minimum SI flowrates. This results in reduced boron addition which maximizes the return to power resulting from the cooldown due to the increased steam flow rate. This is described in detail in Section 3.2.2.11 of Enclosure 2 of the licensee's June 2, 2004, letter. This is conservative and acceptable.

The licensee includes the addition of heat to the reactor coolant due to the cooldown of the reactor coolant below the temperature of the reactor vessel, SG tubes and other metal components of the RCS. The heat addition is minimized (since this results in a higher reactitivity addition and power generation) by assuming conservatively large masses and conservative small heat transfer coefficients.

The licensee discussed the conservatism in the MSLB accident mass and energy release calculations in the June 2, 2004, letter, and in response to the NRC staff's RAIs. The licensee also stated that the modeling of initial conditions, RCS average temperature, and SG water level, are consistent with the conservative intent of the method documented in the NRC staff-approved topical report on MSLB accident mass and energy release calculations. The NRC staff finds that the conservatism in the licensee's analysis of mass and energy release is acceptable.

The MSLB accident mass and energy release calculations have been done using previously approved calculation methods with conservative assumptions and are, therefore, acceptable.

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38 WCAP-8822 (Proprietary) and WCAP-8860 (Non-proprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976.

39 WCAP-8822-S1-P-A (Proprietary) and WCAP 8860-S1-A (Non-proprietary), "Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1976.

3.5.2 MSLB Accident Containment Analysis

A single-node MAAP-DBA model is used to calculate the containment response to an MSLB accident.

The licensee uses the Uchida heat transfer correlation to model heat transfer to structures. As stated above, the Uchida correlation has been used in many NRC-approved containment analyses.

A value of 8 percent is used for the percent of condensate that is revaporized into the containment atmosphere under superheated conditions. This is in accordance with the guidance of NUREG- 0588[40] and is acceptable for this application.

A sufficiently large MSLB will actuate the containment quench spray system. The containment recirculation spray system is not actuated.

Table 4-18 of Enclosure 2 to the licensee's June 2, 2004, submittal for BVPS-1, shows that a peak pressure of 42.6 psig occurs for the 1.4 ft^2 double-ended rupture at 30 percent power with the failure of the main steam check valve. This is a different worst-break than that documented in the UFSAR for the current power level and subatmospheric conditions.

Table 4-19, for BVPS-2, shows that a peak pressure of 39.3 psig occurs for the 1.069 ft^2 double-ended rupture at 30 percent power with failure of one train of quench spray. This is also a different worst-break than documented in the BVPS-2 UFSAR for the current power level and subatmospheric containment conditions.

The difference in results for the containment conversion compared to those in the UFSARs for BVPS-1 and 2 is expected, considering the new analyses are performed at a different power level with different analytical methods.

The licensee discussed the conservatism in the MSLB containment analysis in response to an NRC staff RAI.[41] The licensee also provided a detailed listing of the conservative assumptions made for the important input and modeling parameters in Table 4-3, "Containment Parameter Limits for Design Basis Analysis" in the June 2, 2004, submittal. The NRC staff finds that the conservatism in the MSLB containment analysis is acceptable.

The NRC staff finds the proposed containment conversion acceptable with respect to the MSLB containment analysis since the methods are acceptable (see Section 3.10 of this SE for the evaluation of the MAAP-DBA code) and that the assumptions and input to the calculations are conservative.

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40 Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, A. J. Szukiewicz, NUREG-0588 USNRC, July 1981.

41 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, to USNRC, Response to Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190, Response to RAI 13, February 11, 2005.

3.6 NPSH of Recirculation Spray and Low-Head SI Pumps

The recirculation spray pumps and low-head SI pumps for both units are centrifugal pumps which require adequate available NPSH to perform as assumed in the safety analyses. The sump water temperature is important in determining available NPSH. It is conservative to overestimate the sump water temperature for NPSH calculations.

The licensee proposes to use Westinghouse methods to determine the mass and energy release for the recirculation spray pump NPSH calculations.[42]

A double-ended guillotine break of the hot-leg produces the highest sump water temperature since all SI and recirculation flow must pass through the core where it absorbs decay heat and spills out the break. The break discharge flows to the sump. MAAP-DBA was used to calculate the sump temperature.

The BVPS-1 analysis credits containment accident pressure in calculating available NPSH. The BVPS-2 analysis does not. BVPS-1 NPSH calculations minimize the containment accident pressure. BVPS-2 calculations assume the containment pressure is equal to the vapor pressure corresponding to the sump water temperature. This is conservative since no credit is taken for containment accident pressure and it is consistent with SRP Section 6.2.2, “Containment Heat Removal Systems.”

The LOCA mass and energy release calculations for the NPSH calculations were generated using the NRC-approved methods of WCAP-8264-P-A.[43]

For NPSH calculations, the licensee determines the break flow enthalpy by mixing together the mass and energy released from the two sides of the break before the flow enters the containment. The pressure flash model is then applied to the mixed steam to determine the amount of break fluid that is directed as saturated water to the containment sump or directed as steam to the containment atmosphere. The pressure flash model minimizes the steam released to the containment atmosphere and maximizes the temperature of the water added to the containment sump. Therefore, this method is conservative and acceptable.

The decay heat model used for the NPSH calculations is the more conservative 1971 ANS model. A comparison of this decay heat model with the 1979 ANS model which is used for the mass and energy calculations for input to the containment response calculations is given by the licensee in Table 3.1-12 Part 1 and Part 2. A comparison of the values for both decay heat models shows that for the time of interest for NPSH calculations, the 1971 ANS model is conservative.

Section 4.6 of Enclosure 2 to the licensee’s June 2, 2004, letter, summarizes the different pump configurations of the BVPS-1 and 2 emergency pumps relevant to NPSH considerations:

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[42] Westinghouse Mass and Energy Release data for Containment Design, WCAP-8264-P-A, Revision 1 (Proprietary), WCAP-8312-A (Non-Proprietary), August 1975.

[43] WCAP-8264-P-A, Revision 1 (Proprietary), WCAP-8312-P (Nonproprietary), “Westinghouse Mass and Energy Release Data for Containment Design,” August 1975.

> When evaluating the NPSH limits, the different pump configurations in BVPS-1 and BVPS-2 result in different evaluations for NPSH. In particular, BVPS-1 has a low head safety injection (LHSI) pump which takes suction directly from the containment sump and consequently, must have sufficient NPSH for sustained operation. For BVPS-2, the LHSI pump stops, and the Emergency Core Cooling System (High Head Safety Injection and 6-inch cold-leg injection lines) receives flow from the discharge of two of the four recirculating spray pumps when the system switches over to containment recirculation. BVPS-2 LHSI pump does not take suction from the sump and the two recirculation spray pumps that switch to RCS injection have heat exchangers to cool the water before it enters the RCS.

The licensee states in Section 4.7.2 of Enclosure 2 to the June 2, 2004, letter, that the limiting NPSH is obtained from the medium- and small-break LOCAs. The reason for this is explained in the licensee's May 12, 2005, letter, in response to RAI Question No. 5. The current analyses for both BVPS-1 and 2 conclude that the large-break LOCA is limiting. For the revised analyses, small and intermediate breaks are limiting except for the BVPS-1 inside recirculation spray pumps for which the double-ended hot-leg break is limiting. The licensee attributes this difference to different analysis methods which better model water holdup in containment and a conservative spray pattern assumed in MAAP-DBA which biases more spray to the center of the containment where there is a higher potential for water holdup.

Part of the calculation of available NPSH, is the consideration of the potential blockage of the sump screen by LOCA-generated debris. The licensing basis for BVPS-1 is RG 1.82.[44] The screen head loss for BVPS-2 is based on test data.[45]

For both BVPS-1 and 2, the licensee's proposed RWST level setpoints for realignment from the RWST to the sump are lower than the current values. This results in more of the cooler (relative to the sump) RWST water injected into the containment before initiation of the recirculation phase. However, the level of the RWST must be maintained such that there is available NPSH to the SI pumps. This is limited by the NPSH requirements for the high-head SI (HHSI) pumps for both BVPS-1 and 2. The licensee performed revised NPSH analyses for the LHSI and HHSI pumps to reflect the change in setpoints for both BVPS-1 and 2. The licensee submitted the results of their analyses in their February 11, 2005, letter. The NRC staff reviewed these results and confirmed that the proposed switchover level setpoints for the RWST will give the necessary NPSH for the LHSI and the HHSI pumps.

The minimum available NPSH for the HHSI pumps occurs at the end of the injection phase while the suction supply is from the RWST.[46]

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44 Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Systems," USNRC, June 1974.

45 Letter from J.E. Cross, Dusquesne Light Company, to USNRC, NRC Generic Letter 97-04, January 6, 1998.

46 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, to USNRC, Response to Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190, Response to RAI 16, February 11, 2005.

The minimum height of water above the suction nozzles for the SI pumps is over 10 feet on BVPS-1 and over 28 feet on BVPS-2. The results of analyses to determine the minimum water level at which pumps taking suction from a containment sump will not experience air intrusion are published in RG 1.82, Revision 3.[47] The water levels above the pump suction nozzles for the BVPS-1 and 2 RWSTs exceed the minimum levels specified in RG 1.82, Revision 3.

The NRC staff finds the licensee's proposed RWST level setpoints for realignment from the RWST to the sump acceptable, based on there being sufficient available NPSH and sufficient water level above the suction nozzles so that vortexing and air entrainment will not occur.

The recirculation spray heat exchangers are the only heat exchangers used to remove heat from the containment following a Phase B containment isolation signal. The licensee states that the heat exchanger characteristics chosen for the NPSH analyses represent a range of performance conditions and characteristics that are the most conservative as determined by sensitivity studies.[48] For BVPS-1, the heat exchanger performance characteristics affect both the sump temperature and containment pressure and therefore, the NPSH. For BVPS-2, the NPSH calculation is independent of the heat exchanger characteristics, since the NPSH is determined only by the sump water level and pump flow rate as the containment pressure is assumed equal to the sump water vapor pressure.

It is important that the analyses do not overestimate the heat exchanger performance characteristics. The licensee has described its program to ensure this:[49]

> The heat exchanger monitoring program at both units requires that these heat exchangers are mechanically cleaned and inspected on a regular frequency. The inspection includes visual examination and eddy current testing of the tube integrity. The tube plugging limits are controlled to be within the assumptions of the performance analysis. This method of ensuring recirculation spray heat exchanger heat transfer capability has been accepted by the NRC as an equally effective alternative to testing that satisfies Generic Letter [GL] 89-13 [50]...

The NRC staff finds the treatment of recirculation spray heat exchangers to be acceptable since it is conservative and satisfies regulatory guidance.

The licensee proposes to put a 12-inch drainage port in the biological shield between the reactor cavity and the lower compartment in each unit. This will minimize the water holdup in

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47 Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3, USNRC, November 2003.

48 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, Response to RAI 9, February 11, 2005 .

49 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, Response to RAI 9, February 11, 2005.

50 USNRC Generic Letter 89-13, Supplement 1, Service Water System Problems Affecting Safety Related Equipment, April 4, 1990.

USNRC Generic Letter 89-13, Service Water System Problems Affecting Safety Related Equipment, July 18, 1989.

the reactor cavity and increase the sump water inventory. The licensee has credited this modification in the NPSH calculations.

The NRC staff finds the licensee's determination of NPSH to be acceptable based on conservative estimates of the containment pressure and sump temperature, a surveillance program for the recirculation spray heat exchangers which complies with the guidance of GL 89-13, and compliance with the guidance of SRP Section 6.2.2, "Containment Heat Removal Systems."

3.7 Inadvertent Containment Spray Calculation

The structural design of the BVPS-1 and 2 containments includes a criterion that the minimum internal containment pressure should not be less than 8 psia. The limiting postulated event for this criterion is an inadvertent containment spray actuation from normal operating conditions. The licensee stated that the minimum pressure for this event at containment conversion and EPU conditions is greater than 8 psia.[51] The MAAP-DBA code is not used for this calculation. The minimum containment pressure is calculated using the current method which uses conservative assumptions (e.g., the final containment air temperature is equal to the minimum RWST water temperature) and the ideal gas law.[52] The NRC staff has reviewed the licensee's method and finds it acceptable.

3.8 Equipment Environmental Qualification (EQ) Envelope

The licensee evaluated the effects of the containment conversion on the environmental qualification of equipment both inside and outside containment for both normal operation and DBA conditions. This evaluation included implementation of containment conversion, replacing SGs on BVPS-1 and the EPU.[53] [54]

As a result of the licensee's evaluation, all electric equipment important to safety was found capable of performing its safety functions without hardware modifications, replacement, or further testing. In some cases, for some LOCAs and MSLB accidents, existing equipment profiles were not completely bounded. In these cases, the licensee reviewed equipment specific vendor EQ reports to verify continued equipment operability. The EQ profiles and doses were adjusted in some cases to bound the revised conditions.

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[51] Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, June 2, 2004, Section 4.7.11 of Enclosure 2.

[52] Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, May 12, 2004.

[53] Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, Response to RAI H.11, May 26, 2005.

[54] Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, June 2, 2004, Section 7.11 of Enclosure 2.

The licensee provided tables summarizing the temperature, pressure and radiological impacts of containment conversion, SG replacement and EPU, on equipment covered by the licensee's EQ program.[55]

The NRC staff finds the licensee's evaluation of the proposed changes to the EQ program due to containment conversion, SG replacement and EPU acceptable based on its completeness and compliance with the requirements of 10 CFR 50.49.

3.9 Combustible Gas Monitoring and Control

By letter dated January 28, 2004, the licensee proposed amendments to the BVPS-1 and 2 licenses relocating the TS requirements for hydrogen monitors to the Technical Requirements Manual, thus eliminating the TS requirements associated with the hydrogen analyzers. These changes were approved by the NRC in a letter dated May 19, 2004. These changes are based on a rulemaking which revised the requirements of 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors." The change to an atmospheric containment does not affect the validity of these changes. The background to the revised rule states that combustible gas generated from design-basis and severe accidents was not a significant risk for large dry and subatmospheric containments.[56] The NRC staff therefore, finds the conversion to an atmospheric containment acceptable with respect to combustible gas control.

3.10 MAAP-DBA

The MAAP-DBA computer code is based on the MAAP4 computer code.[57] MAAP4 has not been reviewed and approved by the NRC for licensing calculations. Its use for a specific application on the Donald C. Cook (DC Cook) docket was accepted by the NRC based on conservative assumptions and acceptable comparisons with relevant data and calculations performed by other computer codes. The NRC staff also performed an independent audit calculation for comparison with the DC Cook licensee's results. The audit results were acceptable.[58] [59]

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[55] Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, Response to RAI H.11, May 26, 2005.

[56] Federal Register Vol. 68, No. 179, page 54125.

[57] MAAP4: Modular Accident Analysis Program, User's manual, Electric Power Research Institute (EPRI) report prepared by Fauske and Associates, Inc. (FAI) May 1994.

[58] Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 234 to Facility Operating License No. DPR-58, and Amendment No. 217 to Facility Operating License No. DPR-74; Indiana Michigan Power Company Donald C. Cook Nuclear Plant, Units 1 and 2, Docket. Nos. 50-315 and 50-316; December 1999.

[59] Letter to E.E. Fitzpatrick, Vice President, Indiana Michigan Power Company, from John B. Hickman, Donald C. Cook Nuclear Power Plant, Units 1 and 2 - Issuance of Amendments Re: Ice Weight and Surveillance Requirement, USNRC, January 2, 1998.

Section 9.1 of Enclosure 2 to the licensee's June 2, 2004, letter, states that the MAAP-DBA code was developed to calculate containment response attributes for a spectrum of postulated LOCAs and MSLB accidents for BVPS-1 and 2. Since the BVPS-1 and 2 containments are considered to be atmospheric for this review, the results of this review do not apply without further NRC review to other containment types such as pressure suppression, ice condenser or subatmospheric.

Section 9.1 states that the containment assessments are implemented in a manner consistent with the NRC guidance provided in the SRP.[60] This includes the use of the Tagami and Uchida heat transfer correlations for heat transfer between the containment atmosphere and passive heat sinks for single node calculations. The containment assessments also model behavior not explicitly mentioned in the SRP, especially use of multiple nodes, natural convection heat transfer and the modeling of droplet behavior resulting from the break flow.

Because MAAP-DBA is based on the MAAP4 computer code which contains many models and many options, this review emphasizes those aspects of MAAP-DBA which are judged to have a significant impact on the types of calculations considered for the proposed applications to the BVPS-1 and 2 containment conversion, EPU and SG replacement. This includes those applications of MAAP-DBA shown in table below (along with the other computer codes employed by the licensee for information).

TABLE PROPOSED APPLICATIONS OF THE MAAP-DBA CODE

DBA	Mass and Energy	Containment Response
LBLOCA (< 1 hour)	W March 1979	MAAP-DBA
LBLOCA (> 1 hour)	MAAP-DBA	MAAP-DBA
SBLOCA (Sump mass inventory and NPSH)	MAAP-DBA	MAAP-DBA
MSLB	WCAP-8822; Supplement 1 & Supplement 2	MAAP-DBA

Since the review does not cover explicitly every model in MAAP-DBA, it is important that MAAP-DBA be adequately benchmarked against relevant data and other analysis tools for the intended applications.

A brief description of the MAAP-DBA modeling of engineered safety features is discussed in Section 2.1.4.2 of the licensee's November 24, 2003, submittal. The MAAP-DBA subroutine GENESF is a generalized PWR engineered safeguards model. The licensee states that this model allows flexibility in specifying pump operation and alignment.

60 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition NUREG-0800 (and later revisions) USNRC July 1981.

The MAAP-DBA modeling of heat exchangers is discussed in Section 2.1.4.3 of the licensee's November 24, 2003, submittal. It is based on elementary heat exchanger theory.[61] The subroutine HREXCH is a model of a shell and tube heat exchanger used for ECCS systems and other systems.

The heat transfer to passive heat sinks for multiple node calculations is described in Section 2.1.3 of the licensee's November 24, 2003, submittal. The subroutine HSNKRB manages the calculation of heat transfer to passive containment heat sinks (walls, interior structures, etc.). Natural convection heat transfer coefficients are calculated from the relationships between the Nusselt number and the Raleigh number for horizontal and vertical walls. These natural convection heat transfer correlations are standard text book relationships. As stated earlier, the Tagami and Uchida heat transfer correlations are used for single volume calculations. The Tagami and Uchida correlations have been previously accepted by the NRC in many licensing calculations.

Heat and mass transfer to the containment spray droplets are described in Section 2.1.4.1 of the licensee's November 24, 2003, submittal. The subroutine SPRAY performs these calculations. Diffusion theory is used to calculate the mass transfer to and from the droplet. The mass transfer coefficient is obtained from the heat and mass transfer analogy. This method is commonly used and is acceptable. Spray droplet heat transfer is modeled using equations from Ranz and Marshall[62] which are of the basic form for this type of heat transfer[63] and are acceptable.

In particular, MAAP-DBA will be used to predict the following parameters to ensure compliance with the respective acceptance criteria for BVPS-1 and 2:

- peak containment pressure
- short- and long-term containment temperature
- containment liner temperature
- long-term sump water temperature
- available NPSH for the ECCS and containment spray pumps
- maximum service water outlet temperature for the containment heat removal heat exchanger

The licensee does not propose to use MAAP-DBA for subcompartment analysis or minimum containment pressure analysis done in conjunction with 10 CFR 50.46 LOCA analyses.[64]

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61 Holman, J.P., Heat Transfer, Chapter 10, 4^{th} Edition, New York, McGraw-Hill 1976.

62 Ranz, W. And Marshall, W., Chemical Engineering Progress, Volume 48, 1952.

63 Kazys K. Almenas and Joseph M. Marchello, The Effect of Drop Evaporation Rate on Containemnt Pressure Transients, Nuclear technology, Volume 41, Mid-December 1978.

64 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, to USNRC, Response to Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190, Response to RAIs 21 and 22, February 11, 2005 .

A single node model of the containment is used to calculate the containment pressure, the containment liner temperature, and the post-accident global gas temperature profiles for EQ. The use of the Tagami and Uchida heat transfer models is consistent with the single node containment model. The heat transfer is conservatively increased by a factor of four when calculating the containment liner temperature. This is consistent with NUREG-0588 and the BVPS-1 and 2 licensing basis.

A multi-node calculation is used for the NPSH and sump water temperature. A multi-node calculation gives a more accurate accounting of the distribution of water in the containment. The capability to model multiple nodes interconnected by flow junctions is called a generalized containment model (GCM). Previous versions of MAAP used a fixed node approach.

The containment spray is modeled with a spray droplet diameter of 1000 microns. The acceptability of this droplet size is discussed in Section 3.2.2, “Large-Break LOCA Containment Response,” of this SE.

Peak containment pressure and temperature occur following a LOCA prior to effective operation of the containment sprays. For the MSLB, the quench spray operates for most of the accident and affects the peak conditions. No credit is taken for the recirculation spray in determining the long-term response to the MSLB accident.

The MAAP-DBA containment model (both single node and multiple node) accounts for the presence of water droplets produced by the blowdown into the containment atmosphere. These droplets have a significant effect on the temperature of the containment atmosphere for the MSLB. As stated by the licensee, because of the large surface-to-volume ratio of the droplets, and because of the high water density and the high specific heat of the droplets when compared to steam, a relatively small droplet water mass is sufficient to eliminate superheat from the containment atmosphere. This results in a lower predicted temperature of the containment atmosphere compared to the predicted temperature if droplets were not considered. The licensee assumes that 10 percent of the nonflashed break liquid consists of droplets 100 microns in diameter. The size of the droplets is consistent with the value recommended for use in the GOTHIC (Generation of Thermal Hydraulic Information for Containments) 7.0 computer code[65] based on the work of Ranz and Marshall.[66] This droplet size is also slightly larger than the break droplet size determined in NUREG /CR 1607.[67] The licensee assumed that 10 percent of the nonflashed liquid is droplets based on experiments discussed later in this section. The licensee credits 8 percent revaporization of condensate which is consistent with NUREG-0588.

The qualification of MAAP-DBA is based on several benchmarks. These are discussed in Section 9 of Enclosure 2 to the licensee’s June 2, 2004, letter, and include:

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[65] GOTHIC 7.0, Numerical Applications, Inc., NAI-8907 July 2001.

[66] W. E. Ranz and W. R Marshalll, Jr., Evaporation from Drops, Part I, Chemical Engineering Progress, Vol. 28 No. 3, March 1952.

[67] A. Koestel, et al., “Drop-Size Estimates for a Loss-of-Coolant Accident,” NUREG/CR 1607, USNRC August 1980.

- Comparison with GOTHIC 6.0[68]
- Separate Effects Experiments
- Integral Effects Containment Experiments

3.10.1 Comparison with GOTHIC 6.0

The GOTHIC code series consists of general purpose thermal hydraulic computer programs for the analysis of nuclear reactor containments. GOTHIC was developed for the Electric Power Research Institute (EPRI) by Numerical Applications, Inc. GOTHIC has been extensively validated against data and analytic solutions.[69] It complies with the requirements of 10 CFR Part 50, Appendix B and 10 CFR Part 21. The NRC has previously reviewed the application of GOTHIC 6.0 to an atmospheric PWR containment.[70]

Figures 9-1 to 9-8 of Enclosure 2 to the licensee's June 2, 2004, letter, provide comparisons of MAAP-DBA and GOTHIC for selected LOCA and MSLB accident cases. Table 9-2 of Enclosure 2 to the licensee's June 2, 2004, letter, summarizes some of the modeling assumptions. The assumptions were identical with the exception that GOTHIC included modeling of entrainment of pools and condensate films while MAAP-DBA did not. Table 9-3 of Enclosure 2 to the licensee's June 2, 2004, letter, summarizes these comparisons and is repeated below.

Summary of MAAP-DBA and GOTHIC Comparisons									
Unit	Case	Results of Comparisons							
		LOCA				Main Steamline Break			
		Pressure (psia)		Temperature (EF)		Pressure (psia)		Temperature (EF)	
		MAAP-DBA	GOTHIC	MAAP-DBA	GOTHIC	MAAP-DBA	GOTHIC	MAAP-DBA	GOTHIC
1	8L	57.57	57.41	267.4	266.3	-	-	-	-
2	3L	58.99	58.29	269.7	268.2	-	-	-	-
1	15M	-	-	-	-	56.8	57.8	342.6	341.3
2	16M	-	-	-	-	51.5	52.9	327.1	329.8

The comparisons of MAAP-DBA and GOTHIC results are close and acceptable.

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68 NAI 8907-09, Revision 4, "GOTHIC Containment Analysis Package Qualification Report," Version 6.0, T. George, et al., December 1997.

69 George, Thomas L., "GOTHIC Version 6.0, Containment Analysis Package," December 1997, EPRI RP4444-1.

70 Letter from USNRC to Mark Reddemann, Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3, September 10, 2001.

3.10.2 Separate Effects Tests

The licensee compared experimental data from several separate effects tests with predictions of those data using MAAP-DBA. These comparisons are described in Section 9.4.2 of Enclosure 2 to the licensee's June 2, 2004, letter. The data were taken from the following sources:

- University of Wisconsin flat plate condensation experiments[71]
- PHEBUS FPT0 experiments[72]
- Dehbi condensation experiments[73]
- JAERI spray heat transfer tests[74]
- Kulic spray heat transfer experiments[75]

Table 9-5 of Enclosure 2 to the licensee's June 2, 2004, letter, compares the MAAP-DBA code predictions to the average heat transfer coefficients reported for the Wisconsin flat plate experiments. These experiments measured heat transfer from steam to flat plates cooled on the back side. The data covered a range of temperatures, velocities and air/steam mass ratios. The agreement was close with the exception of Case 6 which was the lowest mass ratio of air to steam. For this case, the result was conservative (predicted heat transfer coefficient was less than the data).

The PHEBUS FPT0 experiment was a variable steam and hydrogen injection into a 10 m^3 steel vessel. The temperature of the walls was controlled to a near constant value to inhibit condensation. Condensation was allowed only on three vertical cylinders that extended from the top inside of the vessel. Vessel pressure and condensation rate were measured. The licensee's predictions and their comparison with data are given in Figures 9-9 and 9-10 of Enclosure 2 to the licensee's June 2, 2004, letter. The figures show close agreement.
Dehbi experimentally measured heat transfer rates between steam-noncondensible gas mixtures and cold surfaces. The effects of pressure, wall length, subcooling and

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71 Huhtiniemi, I.K. et al., Condensation in the Presence of Noncondensible Gases, "Nuclear Engineering and Design," Volume 141 1993.

72 Von der Hardt, et al., "Nuclear safety Research: The Phebus FP Severe Accident Experimental Program," Nuclear Safety, Volume 35, No. 2 July-December 1994.

73 (i) Dehbi, A.A., et al., "The Effects of Noncondensible Gases on Steam Condensation Under Turbulent Natural Convection Conditions," MIT Report MIT-ANP-TR-004 1991a.

(ii) Dehbi, A.A., et al., "Condenstaion Experiments in Steam-Water and Steam-Air-Helium Mixtures Under Turbulent Natural Convection," AIChE Symposium Series, Heat Transfer, Volume 87, pp. 19-28 1991.

74 Kitani, S., "Containment Spray Experiments for Pressure Suppression," Paper Presented at 1st International Conference on Liquid Atomization and Spray Systems, Tokyo, Japan, August 27-31, pp. 355-359 1978.

75 Kulic, E., "An Experimental and Theoretical Study of Simultaneous Heat and Mass Transfer Applied to Steam Dousing," Doctoral Thesis Presented to the Chemical Engineering Department, University of Waterloo, Ontario, Canada 1976.

noncondensible gas fraction were studied. Figure 9-11 of Enclosure 2 to the licensee's June 2, 2004, letter, compares MAAP-DBA predictions with the Debhi data. Reasonable agreement is shown. The agreement improves as the air mass fraction increases. MAAP-DBA predictions were compared with two sets of spray data, the JAERI spray heat transfer tests and the Kulic spray heat transfer experiments. The JAERI tests were a series of pressure suppression spray tests conducted in Japan in a 700 m^3 steel vessel. Subcooled spray water was injected into the vessel containing a saturated air/steam mixture at 3.5 bar. Figures 9-12 and 9-13 of Enclosure 2 to the licensee's June 2, 2004, letter, provide comparisons, respectively, with a single-node and a multi-node representation. The agreement is close in both cases. The single-node representation underpredicts the peak pressure while the multi-node representation is more accurate.

The Kulic tests sprayed water into a 56 m^3 steam-filled vessel. The results are shown in Figure 9-14 of Enclosure 2 to the licensee's June 2, 2004, letter. Test 1 was for the case of no spray; the pressure reduction is due solely to condensation on the walls of the vessel. Test 1A is a single spray nozzle test and Test 1B for a group of five spray nozzles. The licensee predicted the data parametrically for a range of spray droplet diameters. The agreement with the test data is close. The single nozzle test pressure is slightly underpredicted, but the rate of depressurization is close.

Based on these comparisons, the NRC staff concludes that MAAP-DBA adequately predicts condensation and spray heat transfer.

3.10.3 Integral Tests

The licensee compared MAAP-DBA predictions with several integral containment experiments. These are listed in Table 9-6 of Enclosure 2 to the licensee's June 2, 2004, letter, and are repeated in the following table.

Integral Effects Tests Used for MAAP-DBA Containment Response Benchmark		
Benchmark*	Test	Application
1	HDR-V44	Large LOCA
2	HDR-T31.5	Large LOCA
3	NUPEC —7-1	Small LOCA
4	CVTR #3	MSLB w/o Containment Spray
5	CVTR #4,5	MSLB w/ Containment Spray
6	BFMC D-16	Large LOCA

* Benchmark numbers 1,2,3, and 6 are International Standard Problems

The Heissdampfreaktor (HDR) is a decommissioned nuclear reactor used to perform large-scale DBA experiments. The HDR containment does not have the large open space of a domestic large dry containment such as at BVPS-1 and 2. However, it is useful for testing analytical methods such as the MAAP-DBA code.

Test HDR-V44 was used as an international standard problem. HDR-V44 was one of the most severe two-phase blowdown injections, both in terms of the source injection rate, and because the break injection room was one of the smallest of any HDR configuration tested.[76] The licensee used MAAP-DBA with the Tagami correlation (and hence, a single-node model) and the 10 percent airborne water assumption. Figures 9-15 and 9-16 of Enclosure 2 to the licensee's June 2, 2004, letter, compare the MAAP-DBA predictions with data. The pressure is overestimated and the calculated temperature of gas in the containment upper compartment is overestimated. Since, as the licensee points out, the containment is highly compartmentalized, and a single-node model is used, caution is necessary in interpreting the temperature comparison. The licensee compares the calculated (average) gas temperature with the measured gas temperature in the upper compartment, which, according to the licensee, contains 43 percent of the total internal containment volume. Since the temperature in the upper compartment is expected to be higher than the average, the comparison demonstrates that MAAP-DBA, with the Tagami correlation and the 10 percent airborne water assumption, has conservatively calculated the temperature.

Test HDR-T-31.5 was also used as an international standard problem. The injection was similar to HDR-V44, but into a larger volume. The licensee also modeled this test with a single node, using the Tagami correlation and the 10 percent airborne water assumption. Figures 9-17 and 9-18 of Enclosure 2 to the licensee's June 2, 2004, letter, compare the predicted and measured pressure and temperature, respectively. The temperature is again the temperature in the upper compartment. MAAAP-DBA conservatively predicts the pressure and temperature for this test.

NUPEC Test —7-1 was a test in a series conducted to investigate hydrogen distribution phenomena in a 1/4 linearly scaled PWR. The test had two phases: a preheat or vessel conditioning phase in which steam was discharged into the vessel for 3 hours, followed by a helium and steam injection phase. The licensee states that the first phase resembles a design-basis small-break LOCA, and that the second phase simulates a severe accident and is not applicable to design-basis events. The licensee therefore chose to model only the preconditoning phase. Figure 9-19 of Enclosure 2 to the licensee's June 2, 2004, letter, compares the MAAP-DBA predictions with data. Agreement for both pressure and temperature is close early in the test, but deviates at later times. The predictions are conservative with respect to the data. The licensee states that this is due to the underprediction of heat transfer from the gases to the passive heat sinks. Since the licensee uses a multi-node model for the small-break LOCA calculations, this comparison was done with a multi-node model.

A series of DBA simulation tests was conducted in the decommissioned reactor containment building of the Carolinas Virginia Tube Reactor (CVTR) in the late 1960s, as part of an effort to

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[76] Jack Tills, Allen Notafrancesco, and Ken Murata, "An Assessment of CONNTAIN 2.0: A Focus on Containment Thermal Hydraulics (Including Hydrogen Distributions)" SMSAB-02-02 Office of Nuclear Regulatory Research, USNRC July 2002.

develop analytic methods for nuclear power plants.[77] These tests have been used as a reference for validating models in various containment computer codes. In Test #3, steam is injected slightly above the operation deck without containment spray. In Tests #4 and #5, spray was initiated approximately 200 seconds after blowdown and lasted about 12 minutes. The MAAP-DBA predictions are compared with data in Figures 9-20 and 9-21 of Enclosure 2 to the licensee's June 2, 2004, letter. MAAP-DBA overpredicts the containment pressure and temperature. Similar results were obtained in comparisons of these data and the NRC CONTAIN 2.0 code.[78] The overprediction in the case of the CONTAIN 2.0 code was attributed to overestimating the enthalpy of the break flow and the use of natural circulation rather than forced circulation heat transfer.

The licensee used data from Battelle Frankfurt Tests D-15 and D-16[79] to verify containment response using the Tagami correlation and to provide a technical basis for the airborne water fraction of the non-flashed portion of a LOCA blowdown. The licensee's analyses assume 10 percent of the nonflashed blowdown break flow is droplets. Following blowdown, the fraction of airborne water is assumed to be zero and the amount of remaining airborne break discharge droplets decreases due to deposition, spray operation, and evaporation. A single-node model was used for these comparisons. The licensee states that[80]

> The CASP1 [Containment Analysis Standard Problem] (D15) and CASP 2 (D16) experiments performed at the Battelle-Frankfurt model containment show the aerosolization and airborne transport of water during blowdown. Assessment of the results of these two tests indicate values of 49% and 65% for the water masses transported as liquid outside the break room.

Thus, the licensee's assumption that 10 percent of the nonflashed blowdown break is droplets is conservative and acceptable.

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77 Schmitt, R.C., Bingham, G.E., Norberg, J.A., "Simulated Design Basis Accident Tests of the Carolina Virginia Tube Reactor Containment - Final Report," IN-1403, UC-80, Idaho Nuclear Corporation, National reactor Testing Station, Idaho, December 1970.

78 Jack Tills, Allen Notafrancesco, and Ken Murata, "An Assessment of CONNTAIN 2.0: A Focus on Containment Thermal Hydraulics (Including Hydrogen Distributions)" SMSAB-02-02 Office of Nuclear Regulatory Research, USNRC July 2002.

79 Kanzleiter, T., "Investigation of the Phenomena Occurring Within a Multi-Compartment Containment After Rupture of the primary Cooling Circuit in Water Cooled Reactors," Battelle Frankfurt Report RS 50-30-D15-2 1980.

Battelle Frankfurt, 1978, "Investigation of the Phenomena Occurring Within a Multi-Compartment Containment After Rupture of the Primary Cooling Circuit in Water-Cooled Reactors," Battelle Frankfurt Report BF RS 50-30-D15-2, 1978.

80 Letter from Richard G. Mende, Director, Performance Improvement, FirstEnergy Nuclear Operating Company, to USNRC, Response to Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190, Response to RAI 14, February 11, 2005.

The licensee provided a sensitivity study which illustrates that the calculated peak containment pressure is insensitive to airborne water fractions greater than 10 percent. The NRC staff agrees with the results of this sensitivity study.

3.10.4 Conclusion of MAAP-DBA Validation

The licensee has demonstrated that MAAP-DBA compares acceptably well with a widely-used industry containment computer code and with applicable separate effects and integral data. The most significant MAAP-DBA containment models such as those for droplet size and behavior, for break flow and spray flow; for heat and mass transfer; and for heat exchanger performance are consistent with data and accepted industry practice. Therefore, the use of MAAP-DBA for BVPS-1 and 2 atmospheric containment calculations is acceptable.

3.11 TS Changes

As discussed in the introduction to this SE report input, the conversion from a subatmospheric to an atmospheric containment requires changes to the BVPS-1 and 2 TSs. These proposed changes are provided in Attachments A-1 and A-2 to the licensee's June 2, 2004, letter, for BVPS-1 and 2, respectively. These changes are evaluated in this section.

1. Proposed change: The AVs of the containment pressure high (Phase A containment isolation), containment pressure high-high (Phase B containment isolation), and the containment intermediate high-high (Items 1.c, 2.c, 3.b.3, and 4.c) in Table 3.3-3 of TS 3.3.2.1, "Engineered Safety Features Actuation System Instrumentation," are revised. This change applies to the TSs for both units.

Evaluation: The containment pressure high, containment pressure intermediate high-high and containment pressure high-high values assumed in the revised containment analyses were raised from the current values to provide margin between the AVs and the revised containment operating pressure. The following table provides the functions of each of these engineered safety features actuations for each unit.

ESFAS	BVPS-1 Functions	BVPS-2 Functions
Containment Pressure-Hi	SI and FW isolation	SI and FW isolation Containment isolation Phase A
Containment Pressure Hi-Hi	Containment Spray Containment isolation Phase B	Containment Spray Containment isolation Phase B
Containment Pressure Intermediate Hi-Hi	Steamline isolation	Steamline isolation

The acceptability of these changes is justified by two considerations. The first is that the setpoints were derived using acceptable setpoint methods.[81] The second is that the applicable LOCA and MSLB accident analyses demonstrate that all acceptance criteria are satisfied. This is demonstrated by the analyses presented in the licensee's June 2, 2004, letter, and subsequent correspondence and as discussed in this SE.

The setpoint methods were supplemented as described in an October 31, 2005, letter from the licensee.[82] The supplemented methods were approved by the NRC in Amendment Nos. 270 and 152 for BVPS-1 and 2, respectively, dated January 11, 2006.

2. Proposed change: The AVs for Refueling Water Storage Tank Level - Low (BVPS-1) and -Extreme Low (BVPS-2) in Table 3.3-3, "Engineered Safety Feature Actuation System Instrumentation," are revised.

Evaluation: The acceptability of these setpoints is justified by two considerations. They were derived using the acceptable setpoint methods referenced in TS change 1 above, and the applicable containment LOCA analyses demonstrate that all acceptance criteria are satisfied using these setpoints. In addition, these proposed levels are acceptable with respect to NPSH and air entrainment (vortexing) considerations (see Section 3.6 of this SE report input).

3. Proposed change: Table 3.3-3, Item 1.1.d, "Refueling Water Storage Tank Level - Auto QS [quench spray] Flow Reduction," is deleted from the table along with the associated SRs for this function in Table 4.3-2, "Engineered Safety Feature Actuation System Instrumentation - Surveillance Requirements." This change applies only to BVPS-1.

Evaluation: The current BVPS-1 quench spray system includes a restricting flow orifice in parallel with a motor-operated cutback valve installed downstream of each quench spray pump to provide quench spray flow for a longer time period to ensure that containment pressure is maintained subatmospheric following the design-basis LOCA (see Section 6.4.2 of the BVPS-1 UFSAR). Upon receipt of an RWST low-level signal, the motor-operated cutback valve closes and directs water through the flow orifice to provide a reduced flow to the quench spray nozzles. Since this design feature is not required for an atmospheric containment, the licensee did not model it in the revised containment analyses. Since the LOCA analyses demonstrate that all acceptance criteria are satisfied, this change is acceptable.

4. Proposed change: P_a is defined as the peak containment internal pressure related to the design-basis LOCA.[83] The value of P_a is revised in TS 3.6.1.3,

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[81] Westinghouse Setpoint Methodology for Protection Systems, BVPS-Unit 1, Westinghouse Electric Company, WCAP-11419, Revision 4, March 2004.

Westinghouse Setpoint Methodology for Protection Systems, BVPS-Unit 2 Westinghouse Electric Company, WCAP-11366, Revision 6, March 2004.

[82] Letter from Thomas S. Cosgrove, Director, Maintenance, Beaver Valley Power Station, to the U.S. Nuclear Regulatory Commission, October 31, 2005.

[83] 10 CFR Part 50 Appendix J Option B.

"Containment Air Locks," and in SR 4.6.1.3. P_a is also revised in TS 6.1.7, "Containment Leakage Rate Testing Program." The BVPS-1 value of P_a is revised from 40.0 psig to 43.3 psig and the BVPS-2 value is revised from 44.7 psig to 44.9 psig.

Evaluation: The value of P_a is revised to be consistent with the results of the containment LOCA safety analyses for the containment conversion from subatmospheric to atmospheric. Since the safety analyses use acceptable methods and assumptions and yield acceptable results (P_a is less than the design pressure), the values of P_a for both units are acceptable.

5. Proposed change: TS 3.6.1.4, "Internal Pressure," is revised by replacing the air partial pressure requirement with a total air pressure requirement of $12.8 psia and #14.2 psia. This change eliminates the need for TS Figure 3.6-1 which relates the acceptable partial pressure of air in the containment to the river water temperature. This change is consistent with the proposed revision from a subatmospheric containment to an atmospheric containment and applies to the TSs of both units.

Evaluation: The TS containment pressure requirement is made consistent with that for an atmospheric containment and the values are consistent with the input to the containment safety analyses and are, therefore, acceptable.

6. Proposed change: TS 3.6.1.5, "Air Temperature," for the containment average air temperature is revised to replace the existing average air temperature requirement based on TS Figure 3.6-1, which is deleted as part of the change to an atmospheric containment, with an average air temperature range of $70 EF and #105 EF.

Evaluation: This temperature range is consistent with the analysis supporting the change to an atmospheric containment and the proposed change is, therefore, acceptable.

7. Proposed change: TS 3.6.2.3, "Chemical Addition System," is revised by deleting SR 4.6.2.3.d.4 which verifies that one of two operating chemical addition pumps in each subsystem will stop following closure of the BVPS-1 cutback control valve (see proposed TS change 3 above).

Evaluation: As discussed above for proposed TS change 3, the BVPS-1 automatic quench spray flow reduction is not modeled in the revised containment safety analyses. The licensee will defeat this function during implementation of the containment conversion. Since the quench spray flow will no longer be reduced on low RWST level, the associated interlock to reduce the number of operating chemical addition pumps upon closure of the valve will also be defeated. Therefore, SR 4.6.2.3.d.4 will no longer be applicable. Since this change is consistent with the proposed plant design and safety analysis, it is acceptable.

8. Proposed change: TS 3.6.3.1, "Containment Isolation Valves," is revised to delete SRs 4.6.3.1.b and 4.6.3.1.e. Deleting these SRs removes the requirement to cycle each spring- or weight-loaded containment isolation valve and verify that the valve remains closed with <1.2 psid differential pressure

across the valve and opens when the differential pressure is >1.2 psid but <6.0 psid.

Evaluation: These check valves serve a containment isolation function. They provide positive closure in the direction of flow (from atmospheric pressure outside containment to vacuum within the containment).

With the conversion to an atmospheric containment, these SRs are no longer applicable since the valves' function tested by these SRs is no longer necessary, therefore, the removal of the related requirement from the TSs is acceptable. The licensee states that these valves are capable of performing the safety function of preventing leakage from the containment after the conversion to an atmospheric containment without modification (Section 7.9 of Enclosure 2 to the licensee's June 2, 2004, letter).

3.12 <u>Containment Isolation</u>

Section 7.10 of Enclosure 2 to the licensee's June 2, 2004, letter, addresses containment integrity and isolation. The change to an atmospheric containment requires a change in P_a. This is discussed in Section 3.11 of this SE. The change also affects previously granted exemptions to containment isolation GDCs of 10 CFR Part 50, Appendix A, as discussed in Sections 3.12.3 and 3.12.4 below.

3.12.1 Containment Leakage Rate Testing

BVPS-1 and 2 are required to comply with the containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. The testing must be performed at a pressure P_a defined in Appendix J as the calculated peak internal containment internal pressure related to the design-basis LOCA as specified in the TSs.

The licensee's June 2, 2004, letter, states that a Type A (integrated leakage rate test,) must be conducted on the BVPS-1 containment prior to conversion to an atmospheric containment since the proposed value of P_a is greater than the value at which the BVPS-1 containment was last tested (in 1993). In addition, the licensee states that all Type B and C testing conducted at a pressure less than the proposed value of P_a will be performed at the higher P_a value prior to conversion to an atmospheric containment. The NRC staff finds that this satisfies the Type A, B, and C, testing requirements for the BVPS-1 containment and is therefore, acceptable.

The licensee states that the last Type A test for BVPS-2 was conducted in 1993 at a pressure which bounds the proposed value of P_a. Therefore, a Type A test is not required prior to conversion of BVPS-2 to an atmospheric containment. The licensee also states that Type B tests of electrical penetrations are performed at a test pressure which bounds the proposed value of P_a. Thus, those tests remain valid after conversion to an atmospheric containment.

The licensee states that all other Type B and all Type C leakage rate testing will be performed prior to conversion of BVPS-2 to an atmospheric containment.[84] The NRC staff agrees with the licensee's assessment that for the BVPS-2 containment, the 1993 Type A test pressure bounds the proposed value of P_a, and that this Type A test need not be repeated for purposes of converting BVPS-2 to an atmospheric containment. The NRC staff also agrees that the Type B tests on electrical penetrations were performed at test pressures which bound the proposed value of P_a, and that these Type B tests thus need not be repeated for purposes of converting BVPS-2 to an atmospheric containment. The NRC staff finds the licensee's statement that it will perform all other Type B tests and Type C leakage rate testing prior to conversion of BVPS-2 to an atmospheric containment to be acceptable.

3.12.2 Containment Purge Supply and Exhaust and Vacuum Ejector Containment Penetrations

Containment purge supply and exhaust penetrations and vacuum ejector containment penetrations contain resilient seals which, in general, warrant more frequent testing than required by other Type C tests specified in the BVPS-1 and 2 Containment Leakage Rate Testing Program required by the BVPS-1 and 2 TSs. The licensee has previously justified not performing additional testing since leakage in these penetrations in a subatmospheric containment would only result in inleakage and the inleakage would be discovered in a short time due to TS SRs.

The licensee has concluded that additional testing of these penetrations is still not needed because of favorable operating experience and because the environmental conditions remain such that increased seal degradation is not expected. The NRC staff agrees that favorable operating experience and the absence of environmental conditions that could lead to increased seal degradation provides a sufficient basis for not increasing the testing frequency at atmospheric containment operating conditions.

3.12.3 Hydrogen Recombiner Discharge Piping Containment Penetrations

The containment isolation configuration for the BVPS-1 hydrogen recombiner discharge piping (penetrations 87 and 88) does not conform to 10 CFR Part 50, Appendix A, GDC 56, "Primary containment isolation." An NRC letter, dated June 22, 1988, approved an exemption for these penetrations based, in part, on the BVPS-1 containment being subatmospheric. Section 7.10.2 of Enclosure 2 to the licensee's June 2, 2004, letter, states that the exemption is still warranted, based on the fact that these penetrations remain closed and locked in TS Modes 1 through 4 when containment isolation is required and that manual isolation is possible. On this basis, the NRC staff agrees that the exemption is still acceptable.

3.12.4 Recirculation Spray Heat Exchanger River Water Monitor Piping Containment Isolation

The containment isolation configuration for the BVPS-1 recirculation spray heat exchanger river water outlet radiation monitor sample lines (penetrations 83, 84, 85, and 86) does not conform to GDC 57, "Closed system isolation valves." NRC approved an exemption from GDC 57 for

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[84] Letter from L William Pearce, Site Vice President, Beaver Valley Power Station, to USNRC, dated June 2, 2004, Enclosure 2 Section 7.10.2.

these penetrations,[85] [86] [87] based, in part, on the releases being minimized by the subatmospheric pressure in containment during the time required for the operator to manually isolate a heat exchanger with a leak from the recirculation spray system into the river. The licensee states that the exemption should still be valid based on the manner by which the licensee maintains the integrity of the heat exchanger shell and tubes, that manual isolation is possible, and that the increase in radiation release through the radiation monitor sample lines due to an atmospheric containment is not significant. The NRC staff reviewed the basis for the exemption and agrees with the licensee’s position that the exemption is still acceptable for the reasons given by the licensee and the fact that the higher containment pressure would not have a significant effect on the heat exchanger tube leakage in the long term after the recirculation pump flow is terminated, and prior to closure of the manual isolation valve.

3.12.5 GL 96-06[88]

GL 96-06 addressed three issues: susceptibility of containment air cooler cooling water systems to either (1) water hammer, or (2) two-phase flow, and (3) overpressurization of containment piping penetrations due to thermal expansion of fluid between closed isolation valves. The first two issues will be evaluated as part of the NRC staff’s review of the proposed BVPS-1 and 2 EPU. The third issue is addressed here.

The licensee acceptably described the as-found plant condition and the measures taken to ensure overpressurization of containment penetrations would not occur at the current power level in Licensee Event Report 96-009-01.[89]

As part of the proposed BVPS-1 and 2 EPU, the licensee reevaluated the potential to overpressurize containment piping penetrations.[90] The licensee stated that the impact of the increase in containment air temperature does not result in piping pressures or stress values exceeding the values previously evaluated in Licensee Event Report 96-009-01. These values remain bounding at the EPU and containment conversion conditions. The NRC staff therefore, finds the licensee’s response to GL 96-06 with respect to containment isolation remains acceptable under containment conversion and EPU conditions.

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85 Letter from J.D. Sieber, Duquesne Light Company, to USNRC, January 11, 1990.

86 Letter from J.D. Sieber, Duquesne Light Company, to USNRC, April 29, 1990.

87 Letter to J.D. Sieber, Duquesne Light Company, from USNRC, Exemption from General Design Criterion 57 - Beaver Valley Power Station, Unit 1 January 26, 1991.

88 Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,” U.S. Nuclear Regulatory Commission, September 30, 1996.

89 Licensee Event Report (LER) 96-009-01, Containment Penetrations Not in Accordance with the Design Basis, Beaver Valley, Units 1 and 2, January 27, 1997.

90 Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, May 26, 2005.

3.13 Radiological Assessment

Section 5 of Enclosure 2 to the licensee's June 2, 2004, letter, discusses the radiological consequences of the containment conversion. The NRC approved the application of the alternative source term (AST) (10 CFR 50.67 and RG 1.183[91]) for BVPS-1 and 2 for site boundary and control room doses. LOCA doses were also evaluated at the Emergency Response Facility.[92] The license amendments are applicable both prior to and after conversion to atmospheric containments and are applicable to EPU conditions.
Enclosure 2 to the licensee's June 2, 2004, letter, addressed other aspects of long-term post-LOCA containment leakage resulting from atmospheric conversion including: radiological consequences of the containment conversion on normal operation; the long-term post-LOCA radiation effects on equipment EQ; post-LOCA vital access; and iodine loading on accident filters.

Normal Operation Assessment

The licensee states in Section 5.3.1 of Enclosure 2 to the June 2, 2004, letter, that the net effect of atmospheric conversion will be a minor reduction in offsite normal operation gaseous effluent dose. The higher post-containment conversion pressure will result in less effluent from the vacuum pumps, producing less frequent activity release from the containment vacuum system. In turn, this will result in a longer decay time for this effluent. The NRC staff agrees with the licensee's assessment.

Long-Term Post-LOCA

The licensee considered the effect on equipment qualification of changing the licensing basis so that the radiological source term from containment leakage is not terminated at the end of one hour as in the present licensing basis. Credit was taken for operation of the supplemental leakage collection and release system (SLCRS) ventilation system that takes suction from areas contiguous with the containment in minimizing radiation in these areas. The licensee states that the impact of the airborne source due to the containment conversion is minimal and that any increase in mild environments will not cause the zone to change from mild to harsh. The impact of increased activity accumulation on SLCRS and control room filters for both units will increase total integrated dose, but safety-related equipment in the affected EQ zones has been reviewed to ensure that the equipment is qualified to the updated levels.

The licensee conducted a vital access assessment and concluded that both BVPS-1 and 2 remain in compliance with NUREG-0737, Item II.B.2, relative to operator mission doses post-LOCA, considering an atmospheric containment. The licensee also assessed the iodine

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[91] Regulatory Guide 1.183, Alternative Source terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors, USNRC, July 2000.

[92] Letter from USNRC to L. William Pearce Vice President FirstEnergy Nuclear Operating Company Beaver Valley Power Station, "Issuance of Amendment Re: Selective Implementation of Alternative Source Terms and Control Room Habitability Technical Specifications Changes," September 10, 2003.

loading on accident filters and concluded that the values are well below RG 1.52[93] acceptance criteria and are, therefore, acceptable.

Based on the above, the NRC staff finds the licensee's assessment of post-LOCA radiation complete and acceptable since acceptable methods were used and the applicable acceptance criteria were satisfied.

3.14 Station Blackout (SBO)

In Section 7.3 of Enclosure 2 to the licensee's June 2, 2004, letter, the licensee states that the SBO coping capability at the current licensed power level is unaffected by the containment conversion. The effect of SBO at the increased power level was addressed by the licensee.[94] The licensee stated that equipment temperatures remain below the profiles of the existing accident/room heat up analyses and containment isolation remains unaffected by the containment conversion at EPU conditions. The NRC staff, therefore, finds the containment conversion at EPU conditions is acceptable with respect to those aspects of SBO which affect the containment. Other aspects will be addressed as part of the NRC staff EPU review.

3.15 Other Considerations

In Section 6, "Other Considerations," of Enclosure 2 to the licensee's June 2, 2004, letter, the licensee provided justification that various design and licensing requirements will continue to be satisfied should the containment conversion and associated plant equipment changes be implemented at the current power level. These are evaluated in this section of this SE.

3.15.1 UFSAR Accident Analyses

3.15.1.1 LOCA Transients

3.15.1.1.1 Large-Break LOCA

The limiting large-break LOCA in a PWR with respect to peak cladding temperature (PCT) and the other criteria of 10 CFR 50.46 is a break in the cold-leg of the RCS. During the reflood phase, ECCS water boils into steam after coming in contact with the hot core. For the steam to escape the core it has to travel through the hot-legs, into the SG tubes, through the reactor coolant pumps (RCPs), and then out the break in the cold-leg. The increased resistance in the loop tends to reduce the reflood rate in the core. This is referred to as steam binding.

An increase in containment pressure would cause the density of the steam exiting the break to increase and the loop resistance to decrease, ultimately causing the steam pressure in the upper plenum of the vessel to decrease. This pressure and the hydraulic height of the liquid

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93 Regulatory Guide 1.52, Revision 2, Design, Testing and maintenance Criteria for Post- Accident Engineered safety Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants, USNRC, March 1978.

94 Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, October 4, 2004.

level in the core is balanced against the hydraulic height of the liquid in the downcomer. When the upper plenum steam pressure decreases, the liquid level in the core increases to maintain the balance. These effects would cause a reduction in the PCT, cladding oxidation, and hydrogen generation. Higher containment pressure also does not adversely affect the ability to maintain a coolable geometry or long-term cooling. Based on this, the NRC staff agrees with the licensee's conclusion that their current large-break LOCA analyses of record, performed with minimum containment pressures of 8.9 and 9.0 psia for BVPS-1 and 2, respectively, bound the analyses at atmospheric pressure or higher for BVPS-1 and 2.

3.15.1.1.2 10 CFR 50.46 Large-Break Reporting Requirements

The licensee's February 11, 2005, letter, stated that the last large-break LOCA analyses were reviewed and approved by the NRC in 1993 for BVPS-1, and 1987 for BVPS-2. The sum of the absolute values of all accumulated PCT changes since then exceeds 50 ^{o}F for both units. Also, as a result of the conversion to operation at atmospheric pressure for the containment, there will be a 91 ^{o}F decrease in PCT in the large-break LOCA analyses for both BVPS-1 and 2. Per 10 CFR 50.46, the licensee submitted a letter (April 13, 2005) with the proposed schedule for re-analysis which is currently under review by the NRC staff. The licensee proposes to adopt their BELOCA analyses as the new analysis of record for BVPS-1 and 2. The BELOCA analyses were submitted to the NRC by letter dated October 4, 2004, and are currently under review. The licensee also proposed an alternate schedule, should the BELOCA analyses not be approved.

Since the effect of increased containment pressure provides increased margin to the LOCA criteria of 10 CFR 50.46, the approval of this license amendment request is not contingent upon the results of the large-break LOCA re-analysis review, nor upon the approval of the BVPS-1 and 2 BELOCA submittal or any other contingent submittal should the BELOCA analyses not be approved.

3.15.1.1.3 Small-Break LOCA

BVPS-1 and 2 small-break LOCA analyses are performed using the Westinghouse NOTRUMP evaluation methodology.[95] This methodology does not use containment backpressure in the model since the break flow is at critical flow conditions for most of the transient. The small-break LOCA analyses are unaffected by the change in containment pressure. Therefore, the NRC staff finds the change to atmospheric containment pressure acceptable with respect to the small-break LOCA analyses.

3.15.1.1.4 10 CFR 50.46 Small-Break Reporting Requirements

The licensee's February 11, 2005, letter, stated that the last small-break LOCA analyses were reviewed and approved by the NRC in 1993 for BVPS-1, and 1987 for BVPS-2. The sum of the absolute values of all accumulated PCT changes since then exceeds 50 ^{o}F for both units. Per

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[95] Rupprecht, S. D., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (Proprietary) 1986.

10 CFR 50.46, the licensee submitted a letter[96] with the proposed schedule for re-analysis which is currently under review by the NRC staff. The licensee proposes to adopt their small-break LOCA analyses as the new analyses of record for BVPS-1 and 2. The small-break LOCA analyses were submitted to the NRC by letter dated October 4, 2004, and are currently under review. The licensee also proposed an alternate schedule, should the small-break LOCA analyses not be approved.

Since the increased containment pressure proposed by the licensee does not affect the small-break LOCA analyses, the approval of this license amendment request is not contingent upon the results of the small-break LOCA re-analysis review schedule. Neither does such approval depend upon the approval of the BVPS-1 and 2 small-break LOCA analyses or any other contingent submittal should the small-break LOCA analyses not be approved.

3.15.1.1.5 SG Tube Rupture

The licensee's June 2, 2004, letter, stated that containment pressure is not used as an input to the SG tube rupture (SGTR) analysis and therefore, the SGTR event is not adversely impacted by the increase in containment pressure. The NRC staff agrees with this conclusion.

3.15.1.2 Non-LOCA Transients

The licensee uses containment pressure as an input for two non-LOCA events, the MSLB and the main feedline break.

3.15.1.2.1 MSLB

An MSLB causes an increase in steam flow which decreases with time as the steam pressure decreases. The increased steam flow causes increased energy removal from the reactor coolant system and results in a reduction of coolant temperature and pressure. Due to the assumed negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. The core reactivity increase causes a decrease in shutdown margin and may cause a power level increase. Analysis of the transient following an MSLB is sensitive to the fluid discharge rate at the break. Past experience generally shows that the worst-break is that which results in the maximum cooldown rate. For the maximum break size, the steam flow out the break would be choked. Increasing the containment pressure would have no effect on this analysis. The NRC staff finds that the proposed increase in containment pressure would result in no negative safety impacts on this event for both BVPS-1 and 2. In addition, since the containment pressure has no effect on the MSLB accident analysis, operation with the containment pressure subatmospheric is also acceptable.

3.15.1.2.2 Feedline Break

Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy

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96 Letter from L.W. Pearce,. Vice President, FirstEnergy Nuclear Operating Company to USNRC, April 13, 2005.

discharge through the break) or an RCS heatup by reducing feedwater flow to the affected SG. The MSLB analyses bounds a potential RCS cooldown resulting from a feedwater line break for containment pressure analyses. A postulated break upstream of the feedline check valve is bounded by the loss of feedwater analysis. If the break is postulated in a feedline between the check valve and the SG, fluid from the SG may also be discharged through the break. Therefore, only the RCS heatup effects where fluid from the SG is also discharged out of the break, are evaluated for a main feedwater line break.

For both BVPS-1 and 2, the break release model for the main feedline break event assumes a constant containment backpressure. Generally at higher SG pressures, the break flow is independent of the containment pressure since the flow is choked; therefore the change in containment pressure would have no effect on the main feedwater line break event. For lower SG pressures, a higher containment pressure may have a slight benefit due to a lower release flow.

AFW is used to mitigate the consequences of the feedline break event. In the current BVPS-1 and 2 main feedline break analyses, all AFW flow is assumed to be lost to the broken feedline until operator action isolates flow to this path. Following isolation, flow to the intact SG commences. The licensee calculates flow to the intact SG based on the pressure difference between the SG and the inlet of the AFW pump. The licensee assumes the most conservative condition which is that the pressure difference across the AFW pump is at a maximum giving the lowest AFW flow. The maximum SG pressure is at the SG safety valve set pressure. This is the pressure that is assumed in the safety analyses and is therefore not affected by containment pressure. The NRC staff finds that the proposed increase in containment pressure would result in no negative safety impacts on this event for both BVPS-1 and 2. Also, for the same reason, operation at the current power level with the containment conversion to atmospheric pressure is acceptable.

3.15.1.2.3 ATWS

In the generic studies that make up the licensing basis for BVPS-1 and 2, Westinghouse determined that the rates of mass and energy release for ATWS transients are significantly lower than for a LOCA event (Figure D-2, 1974 report[97 98]). The NRC staff agrees that this would be the case with the increase in containment operating pressure at the current power level.

3.15.1.2.4 Natural Circulation

The licensee's June 2, 2004, letter, states that conversion to atmospheric containment pressure does not adversely impact the natural circulation flow and boron mixing capability of the reactor coolant system for BVPS-1 and 2, since natural circulation is independent of containment pressure. The NRC staff agrees with this conclusion.

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97 Burnett, T. W. T., et al., "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

98 Anderson, T. M. (Westinghouse) to S. H. Hanauer (USNRC), "ATWS Submittal," NS-TMA-2182, December 1979.

3.15.1.3 Initial Condition Uncertainties

Enclosure 2 to the licensee's June 2, 2004, letter, states that the changes associated with the conversion to atmospheric containment conversion have no adverse impact on the initial condition uncertainties used as inputs to the UFSAR accident analyses. With the exception of the containment pressure and RWST level setpoints, the changes associated with conversion to atmospheric containment will not impact the reactor trip system and engineered safety features actuation system (RTS/ESFAS) setpoints. The NRC staff agrees with this conclusion. As discussed in ths section, the changes to the containment pressure and RWST level set points are acceptable at the current power level.

3.15.1.4 ECCS

The ECCS consists of the HHSI pumps, RWST, LHSI pumps, and the SI accumulators. The ECCS operates in the injection phase and the recirculation phase. The injection phase provides borated water from the RWST to the reactor vessel. The recirculation phase provides long-term post-accident cooling by recirculating water from the containment sump.

The licensee states that[99]

> With an atmospheric containment and a lower minimum allowable containment temperature, the ECCS will operate essentially the same as with a subatmospheric containment. System functional and performance requirements have not been modified with the exception of an increase in the injection water temperature (65 EF[100] vs. 55 EF assumed in the UFSARs) and a lower RWST level setpoint....
>
> Key system parameters including pump performance requirements, NPSH, and effects of temperature changes on piping and components have been reviewed to assure continued safe operation of the system to perform its design function.

ECCS pump NPSH is addressed in Section 3.6 of this SE and has been found to be acceptable for containment conversion to an atmospheric containment and EPU conditions. With containment conversion but without the EPU, sump temperatures would be lower and there would be more NPSH margin. The calculation of the maximum service water outlet temperature, which is used for piping analyses, is described in Table 4-3 of Enclosure 2 to the licensee's June 2, 2004, letter. These calculations are done conservatively and are therefore, acceptable. See Section 3.15.8 of this SE for further discussion of piping systems.

Based on the above discussions, the NRC staff finds the design of the ECCS acceptable for containment conversion.

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[99] Letter from L. William Pearce, Site Vice President, Beaver Valley Power Station, to USNRC, dated June 2, 2004, Enclosure 2 Section 6.3.11.

[100] Letter from L. William Pearce, Site Vice President, Beaver Valley Power Station, to USNRC, dated June 2, 2004, Enclosure 2 Table 4-3.

3.15.1.5 UFSAR Accident Analyses Conclusions

Based on the above considerations, the NRC staff finds that increasing containment pressure will have no adverse safety effects on LOCA transients, SG tube rupture, non-LOCA transients, ATWS, natural circulation, initial condition uncertainties, and the ECCS.

3.15.2 NSSS System Components and Nuclear Fuel

The licensee states that the containment conversion will have no adverse effect on the following NSSS components located within the containment:

- reactor vessel
- SGs
- RCPs
- control rod drive mechanisms
- loop stop isolation valves
- NSSS auxiliary equipment

Based on a review of the design of these systems, the NRC staff agrees that the relatively small change in containment operating pressure will not adversely impact the operation of these NSSS components. The effect of the proposed EPU on these components will be discussed in the NRC staff's SE on the EPU. Based on the above considerations, the NRC staff also finds that increasing containment pressure will have no adverse safety effects on the NSSS and its components or the nuclear fuel.

3.15.3 Safety-Related Valves

Sections 7.6, 7.7, 7.8, and 7.9 of Enclosure 2 to the licensee's June 2, 2004, letter, address motor-operated valves, air-operated valves, relief valves, and check valves.

The licensee states that the containment conversion will not change any program controls, or existing licensee commitments with respect to the BVPS-1 and 2 motor-operated valve program. The torque and thrust requirements are based on a maximum containment pressure equal to the 45 psig containment design pressure. The safety analyses for both the current containment design and the containment conversion show that the pressure remains below this value for the postulated DBAs. Postulated pipe breaks assume that the downstream side of the valve is exposed to atmospheric pressure. Therefore, subatmospheric conditions are conservatively bounded.

The licensee states that air-operated valve characteristics are determined by test. The testing done by valve manufacturers for BVPS-1 and 2 and the testing done in-situ are performed at atmospheric pressure. This is conservative relative to subatmospheric pressure.

Therefore, operation of both BVPS-1 and 2 at the current power level with an atmospheric containment is acceptable. Acceptability of the operation of safety-related valves at EPU conditions will be discussed in the NRC staff's SE of the licensee's proposed EPU.

3.15.4 Supplementary Leak Collection and Release System (SLCRS)

Because SLCRS operates outside containment, the containment conversion will not affect its operation. The operation of SLCRS at EPU conditions will be considered in the NRC staff's ongoing review of the licensee's EPU submittal.

3.15.5 Quench Spray System

The licensee states that::

> With an atmospheric containment, the [quench spray] system will operate essentially the same as with a subatmospheric containment. System functional and performance requirements have not been modified with the exception of the proposed deletion of a cutback system at BVPS-1, an increase in the maximum spray temperature and a higher CIB [containment isolation phase B] set point.

The deletion of the quench spray cutback system is discussed in Section 3.11 of this SE. The higher quench spray temperature and the higher CIB set point are acceptable based on the acceptable results of the safety analyses performed by the licensee.

3.15.6 Recirculation Spray System

The licensee states that:

> No physical changes are being made to the [recirculation spray] system to support atmospheric containment or lower minimum containment temperature. [The licensee proposes changing the lower bound initial containment air temperature from 75 EF to 70 EF in TS Limiting Condition for Operation 3.6.1.5 for both units].

The NRC staff finds, based on its engineering judgement, that the relatively small change in containment pressure and ambient temperature will not have an adverse effect on the recirculation spray system. The NRC staff agrees that no changes need be made to the recirculation spray system.

3.15.7 Chemical Addition System

The licensee stated that key system parameters had been reviewed to assure successful operation of the system to perform its design function at atmospheric containment operating conditions. The chemical addition system will continue to perform its design function following conversion of the containments to atmospheric conditions because the design function is not dependent on containment pressure. The NRC staff agrees with the licensee's assessment on this point.

3.15.8 Piping Systems

The licensee states that:[101]

> For the containment atmospheric conversion, piping, pipe supports, and equipment nozzle and related support systems, remain within allowable stress limits.

The NRC staff agrees with the licensee's assessment that following containment conversion to atmospheric conditions, the piping, piping supports and equipment nozzle and related support systems will remain within the allowable stress limits. Acceptability of the operation of piping systems at EPU conditions will be discussed in the NRC staff's SE of the licensee's proposed EPU.

3.15.9 River Water System

The licensee states that:[102]

> With an atmospheric containment, the [River Water] system will operate the same as with a subatmospheric containment. System functional requirements have not been modified. Key system parameters including pump performance requirements and the effects of temperature changes on piping and components have been reviewed to assure successful operation of the system to perform its design function.

Because the river water system will be unaffected by conversion to atmospheric containment operating conditions, this is acceptable to the NRC staff. Acceptability of the operation of the river water system at EPU conditions will be discussed in the NRC staff's SE of the licensee's proposed EPU.

3.15.10 Service Water System

The licensee states that:[103]

> With an atmospheric containment, the [service water] system will operate the same as with a subatmospheric containment. System functional requirements have not been modified. Key system parameters including pump performance requirements and the effects of temperature changes on piping and components

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[101] Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, October 4, 2004, Section 6.3.12.

[102] Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, October 4, 2004, Section 6.3.13.

[103] Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, October 4, 2004, Section 6.3.14.

> have been reviewed to assure successful operation of the system to perform its design function.

Because the service water system will not be adversely affected by the conversion of the containments to atmospheric operating conditions, this is acceptable to the NRC staff. Acceptability of the operation of the service water system at EPU conditions will be discussed in the NRC staff's SE of the licensee's proposed EPU.

3.15.11 Main Feedwater System

The licensee states that:[104]

> With an atmospheric containment the main feedwater system will operate essentially the same as with a subatmospheric containment.

Based on its MSLB analysis, set forth in SE Sections 3.5 and 3.15.1.2.1, the NRC staff agrees with the above-quoted licensee statement. The NRC staff notes that in order to maintain the containment pressure less than the acceptance criterion for the MSLB accident, the licensee plans to add, pursuant to 10 CFR 50.59, a new, fast-acting feedwater isolation valve to each of the three lines supplying main feedwater to the BVPS-1 SGs. The addition of these valves will make BVPS-1 similar to the existing valve arrangement at BVPS-2 in this regard.

3.15.12 AFW System

The licensee states that:[105]

> With an atmospheric containment and the current licensed power level, the auxiliary feedwater system will operate essentially the same as with subatmospheric conditions. However, reduced auxiliary feedwater flow to the depressurized steam generator is required to prevent containment overpressurization following a main steamline break accident.

In order to limit the AFW flow to the depressurized SG, the licensee will install cavitating venturis in each of the BVPS-1 AFW supply lines to each SG. This is similar to the existing BVPS-2 arrangement. This is discussed in the Introduction of this SE. Installation of these cavitating venturis in the BVPS-1 AFW system is acceptable for the containment accident analyses based on the acceptable results of the BVPS-1 MSLB accident analyses.

The acceptability of installation of these cavitating venturis in the BVPS-1 AFW system on pipe breaks outside containment will be discussed in the NRC staff's SE of the licensee's proposed EPU.

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[104] Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, October 4, 2004, Section 6.3.15.

[105] Letter from L. William Pearce, Site Vice President, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, to USNRC, October 4, 2004, Section 6.3.16.

3.15.13 Instrumentation

The licensee has reviewed the impact of operation with atmospheric containment pressure on pressure and temperature instrumentation located within containment. The licensee states that a review of this instrumentation has identified instrumentation that is acceptable as-is; to be replaced; or to be rescaled to accommodate operation at atmospheric conditions.

The licensee has assessed the impact of atmospheric containment pressure on RTS and ESFAS instrumentation and has concluded that the intended safety functions will be performed within acceptable limits.

In addition, the ESFAS function for automatic transfer from injection to recirculation using the refueling water storage tank level-low channels, will be rescaled to new setpoint values. Since these values have been used in the safety analyses with acceptable results, this is acceptable.

The licensee states in Section 6.1.6 of Enclosure 2 to the June 2, 2004, letter, that:

> the changes associated with the containment conversion listed in Section 6.1 [UFSAR Accident Analyses] have no adverse impact on the initial condition uncertainties (i.e., reactor power and RCS flow, temperature, and pressure uncertainties) used as input to the UFSAR accident analyses. Similarly, the changes associated with containment conversion will not impact the RTS/ESFAS [reactor trip system/engineered safety features actuation system] setpoints except for the containment pressure and the RWST level setpoints that have been revised as part of the containment conversion.

The NRC staff has previously found above in SE Section 3.11 that the licensee's changes to the containment pressure and RWST level setpoints are acceptable. The NRC staff agrees with the licensee that the other changes listed in Section 6.1.6 of Enclosure 2 to the June 2, 2004, letter will not impact the initial condition uncertainties because they are independent of containment pressure.

4.0 Conclusion

The licensee has proposed to convert the subatmospheric containments of BVPS-1 and 2 to atmospheric containments. This proposal was accompanied by appropriate changes to the TSs of both units. To support these changes, the licensee provided supporting analyses. These analyses were performed at a power level 8 percent above the current rated thermal power in order to accommodate a planned EPU of both units. The analysis also included the replacement of the BVPS-1 SGs. The analyses used a combination of previously NRC-approved Westinghouse methods and the MAAP-DBA computer code. The licensee also demonstrated that operation at the current power level is acceptable with an atmospheric containment.

The NRC staff has determined that the licensee's proposal is acceptable because safety-related structures, systems, and components will continue to perform their design-basis function following the conversion of the BVPS-1 and 2 containments to atmospheric operating conditions. MAAP-DBA is appropriate for BVPS-1 and 2 containment analyses except for subcompartment analyses and minimum containment pressure 10 CFR 50.46 LOCA analyses for which the licensee is not proposing to use MAAP-DBA. The licensee's safety analyses are complete, conservative and follow the guidance of the appropriate sections of the SRP except where additional justification has been provided. The proposed containment conversion complies with the appropriate GDCs and other applicable regulations discussed in this SE.

Therefore, the licensee's proposal to operate BVPS-1 and 2 with a containment internal air pressure between 12.8 psia and 14.2 psia and at a rated thermal power level of either 2689 MWt or 2900 MWt is acceptable.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (69 FR 43462). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 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: R. Lobel
V. Klein

Date: February 6, 2006

Appendix 1: Mass and Energy Release Confirmatory Calculations For Beaver Valley Power Station To Support License Amendment for Conversion to Atmospheric Containment Long-term Mass and Energy Release: Verification of Separate Mass and Energy Release Calculations for Long-term (> 1 Hour)

Appendix 2: Mass and Energy Release Confirmatory Calculations For Beaver Valley Power Station To Support License Amendment for Conversion to Atmospheric Containment: Short-term Mass and Energy Release

APPENDIX 1

MASS AND ENERGY RELEASE CONFIRMATORY CALCULATIONS

FOR BEAVER VALLEY POWER STATION

TO SUPPORT LICENSE AMENDMENT FOR CONVERSION TO

ATMOSPHERIC CONTAINMENT

LONG-TERM MASS AND ENERGY RELEASE:

VERIFICATION OF SEPARATE MASS AND ENERGY RELEASE CALCULATIONS FOR

LONG-TERM (> 1 HOUR)

MASS AND ENERGY RELEASE CONFIRMATORY CALCULATIONS

FOR BEAVER VALLEY POWER STATION

TO SUPPORT LICENSE AMENDMENT FOR CONVERSION TO

ATMOSPHERIC CONTAINMENT

LONG-TERM MASS AND ENERGY RELEASE:

VERIFICATION OF SEPARATE MASS AND ENERGY RELEASE CALCULATIONS FOR

LONG-TERM (> 1 HOUR)

1.0 INTRODUCTION

By letter dated June 2, 2004,[106] FirstEnergy Nuclear Operating Company (FENOC, the licensee) submitted a license amendment request for Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2). FENOC requested to convert its containment operating conditions from sub-atmospheric to atmospheric pressure. The licensee also proposed changes to their current licensing basis methodology. The proposed change would allow the licensee to use the modular accident analysis program-design basis accident (MAAP-DBA) code to evaluate the energy release to the containment after 3600 seconds (1 hour) after the initiating event. This is a deviation from the approved licensing basis that uses the Westinghouse March 1979 loss-of-coolant accident mass and energy release model[107] to calculate mass and energy releases up until 6 hours after the initiating event. The licensee is still using the approved methodology to determine the mass release rates. In addition, the licensee's calculations have been performed at extended power uprate conditions.

In response to the Nuclear Regulatory Commission (NRC) staff's requests for additional information (RAIs), the licensee supplemented their application by letters dated February 11,[108]

106 Letter from L.W. Pearce (FirstEnergy Nuclear Operating Company) to U.S. Nuclear Regulatory Commission "Beaver Valley Power Station, Unit No. 1 and No. 2 BV-1 Docket No. 50-334, License No. DPR-66 BV-2 Docket No. 50-412, License No. NPF-73 License Amendment Request Nos. 317 and 190," June 2, 2004.

107 WCAP-10325-P-A, (Proprietary), WCAP-10326-A (Nonproprietary), "Westinghouse LOCA Mass & Energy Release Model for Containment Design - March 1979 Version," May 1983.

108 Letter from R.G. Mende (First Energy Nuclear Operating Company) to U.S. Nuclear Regulatory Commission "Beaver Valley Power Station, Unit No. 1 and No. 2; BV-1 Docket No. 50-334, License No. DPR-66; BV-2 Docket No. 50-412, License No. NPF-73; Response to a Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190" February 11, 2005.

April 13, [109] May 12,[110] and November 14, 2005.[111] The NRC staff also performed an audit of the licensee's calculations on November 7, 8, and 9, 2005.

The NRC staff performed a confirmatory calculation of the double-ended pump suction break because it yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the reactor coolant system in calculating the releases to containment. The NRC staff used RELAP5[112] to evaluate the licensee's results for this event for long-term mass and energy release.

2.0 ASSUMPTIONS

The NRC staff used the decay heat curve on page 3-55 of Enclosure 2 to the licensee's June 2, 2004, letter, and emergency core cooling system injection rates from page 3 of the licensee's May 12, 2005, letter. The NRC staff assumed switchover to sump recirculation at 2900 seconds after the initiating event and assumed an injection rate of 425 lbm/s. The NRC staff assumed a sump temperature of 186 ^{o}F that decreases with time to 148 ^{o}F and a containment back pressure which starts at 60 psia at the initiation of the event and decreases to 14.7 psia, 2500 seconds later. All of these inputs are consistent with the values used in the licensee's submittal.

3.0 RESULTS

The results of the NRC staff's RELAP5 calculation follow. Figures 1, 2, and 3 show the comparisons of the NRC staff and licensee calculations for the integrated energy release, long-term mass release rate, and long-term energy release rate, respectively.

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[109] Letter from L.W. Pearce (First Energy Nuclear Operating Company) to U.S. Nuclear Regulatory Commission "Beaver Valley Power Station, Unit No. 1 and No. 2 BV-1 Docket No. 50-334, License No. DPR-66 BV-2 Docket No. 50-412, License No. NPF-73 10 CFR 50.46(a)(3)(ii) Schedule for Reanalysis," April 13, 2005.

[110] Letter from L.W. Pearce (First Energy Nuclear Operating Company) to U.S. Nuclear Regulatory Commission "Beaver Valley Power Station, Unit No. 1 and No. 2; BV-1 Docket No. 50-334, License No. DPR-66; BV-2 Docket No. 50-412, License No. NPF-73; Response to a Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190," May 12, 2005.

[111] Letter from James H. Lash, Site Vice President, Beaver Valley Power Station, to U.S. Nuclear Regulatory Commission, Supplement to Responses to request for additional Information dated March 28, 2005, in Support of License Amendment Request Nos. 317 and 190, November 14, 2005.

[112] "RELAP5/MOD3 Code Manual," NUREG/CR-5535 US Nuclear Regulatory Commission June 1999.

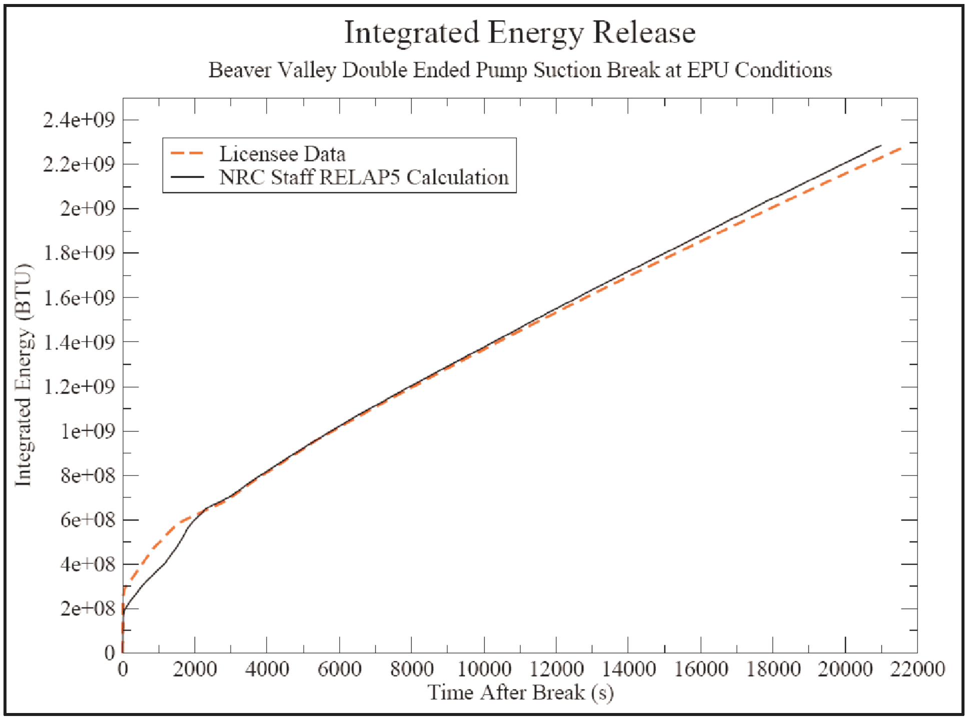

Figure 1: Integrated Energy Release

Figure 1 shows that the two calculations are in agreement for the integrated energy release. At the beginning of the transient up to about 2000 seconds, the licensee's calculation is higher than the the NRC staff's. This is most likely due to the modeling convention taken by the licensee in which the release of the metal mass energy is assumed to occur very early in the transient while the NRC staff's calculation assumes a more gradual release. At 6 hours the NRC staff's calculation is about 2 percent higher than the licensee's.

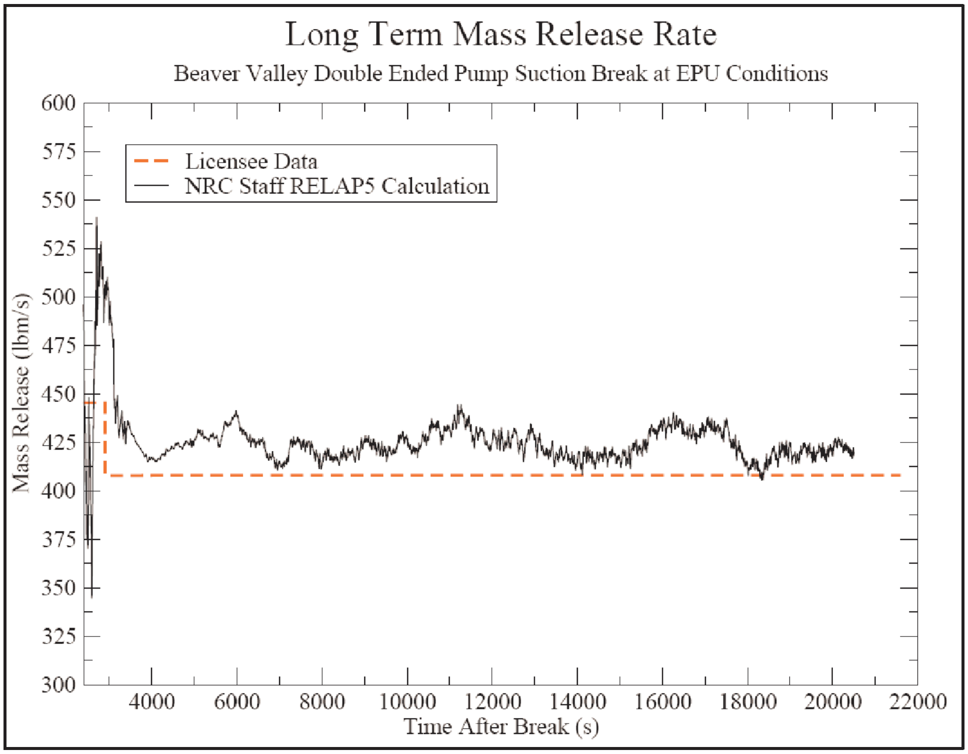

Figure 2: Long Term Mass Release Rate

Figure 2 shows relatively close agreement between the licensee and the NRC staff's calculations. The differences are most likely due to the assumptions used in the calculation of mass release. The licensee used the last value of 408 lbm/sec from the calculation in Enclosure 2 to the licensee's June 2, 2004, letter, which ended at 3600s. When calculating the recirculation injection rate, the NRC staff converted the value given in the licensee's April 13, 2005, letter, from gallons-per-minute to pound-mass-per-second using 62 lbm/ft^3 as the density of water. Given that the sump conditions change over time, the density of water could be lower. This could cause the NRC staff's calculated mass release rate to be lower by as much as 10-20 lbm/s.

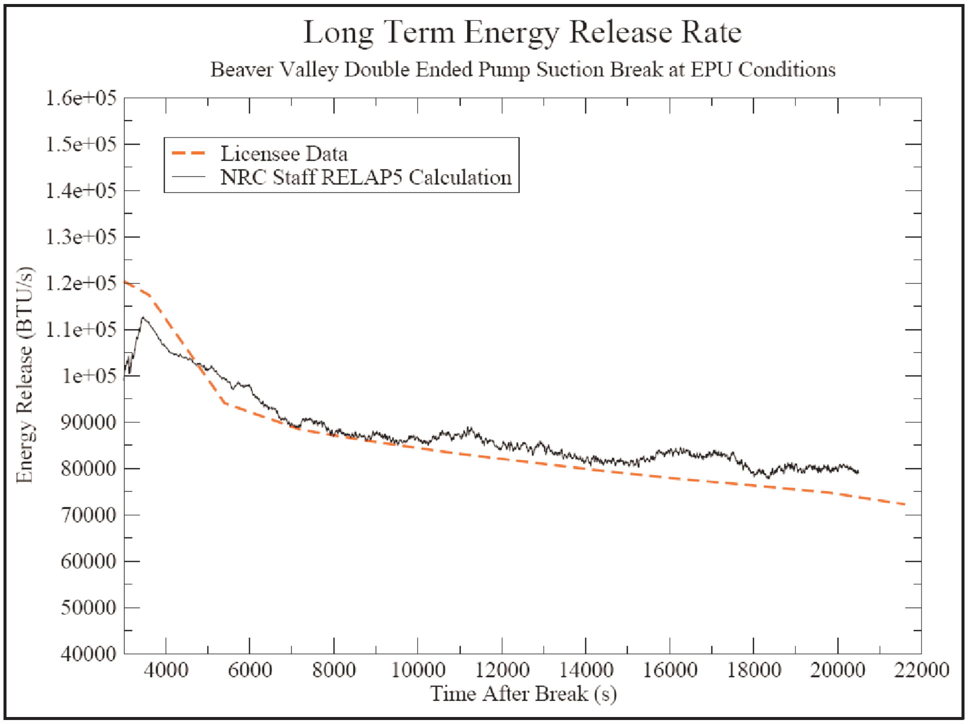

Figure 3: Long Term Energy Release Rate

Figure 3 shows good agreement between the licensee and the NRC staff's calculations. At about 16,000 seconds the NRC staff's calculation is about 4 percent higher than that of the licensee's, however, this may be due to the higher mass release calculated by the NRC staff which is also about 4 percent higher.

4.0 CONCLUSION

The NRC staff concludes that the MAAP-DBA methodology used to determine long-term energy release rate is acceptable for BVPS-1 and 2 based upon agreement with the NRC staff's RELAP5 calculation.

APPENDIX 2

MASS AND ENERGY RELEASE CONFIRMATORY CALCULATIONS

FOR BEAVER VALLEY POWER STATION

TO SUPPORT LICENSE AMENDMENT FOR CONVERSION TO

ATMOSPHERIC CONTAINMENT:

SHORT-TERM MASS AND ENERGY RELEASE

MASS AND ENERGY RELEASE CONFIRMATORY CALCULATIONS

FOR BEAVER VALLEY POWER STATION

TO SUPPORT LICENSE AMENDMENT FOR CONVERSION TO

ATMOSPHERIC CONTAINMENT:

SHORT-TERM MASS AND ENERGY RELEASE

1.0 INTRODUCTION

By letter dated, June 2, 2004,[113] FirstEnergy Nuclear Operating Company (FENOC, the licensee) submitted a license amendment request for Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2). FENOC requested to convert its containment operating conditions from sub-atmospheric to atmospheric pressure. As part of its analyses to support this request, the licensee calculated the mass and energy release rates for the double-ended hot-leg (DEHL) break and the double-ended pump-suction (DEPS) break. The Nuclear Regulatory Commission (NRC) staff performed confirmatory calculations of the BVPS-1 mass and energy release for both events using the thermal-hydraulic code RELAP5.[114]

2.0 MODELING PARAMETERS

The NRC staff used the extended power uprate conditions quoted in the licensee's June 2, 2004, submittal, of 2917.4 megawatt thermal and the 1971 American Nuclear Society Standard Decay Heat Curve. The RELAP5 analysis employed the Henry-Fauske critical flow model.[115]

3.0 RESULTS

3.1 DEHL Break

The DEHL break location yields the highest blowdown mass and energy release rates. The licensee calculates the mass and energy releases for the blowdown phase of the hot-leg break and uses this in the containment peak pressure and temperature response calculation. The results of the NRC staff's mass and energy release calculation for the DEHL break are summarized in Figures 4 through 9. Figure 4 shows the mass release rate as a function of time for the blowdown phase of the DEHL break compared to the licensee's calculation. The x-axis shows time after the break in seconds, the y-axis presents the sum of mass release rates from

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[113] Letter from L. W. Pearce (FirstEnergy Nuclear Operating Company) to U.S. Nuclear Regulatory Commission "Beaver Valley Power Station, Unit No. 1 and No. 2 BV-1 Docket No. 50-334, License No. DPR-66 BV-2 Docket No. 50-412, License No. NPF-73 License Amendment Request Nos. 317 and 190," June 2, 2004.

[114] "RELAP5/MOD3 Code Manual," NUREG/CR-5535 US Nuclear Regulatory Commission June 1999.

[115] R. E. Henry and H. K. Fauske, "The Two-Phase Critical Flow of One-Component Mixtures in Nozzles, Orifices, and Short Tubes," Transactions of ASME, Journal of Heat Transfer, 93, 1971, pp. 179-187.

the vessel side and the pump side of the break in the discharge leg. The NRC staff's calculation shows close agreement with the licensee's analysis.

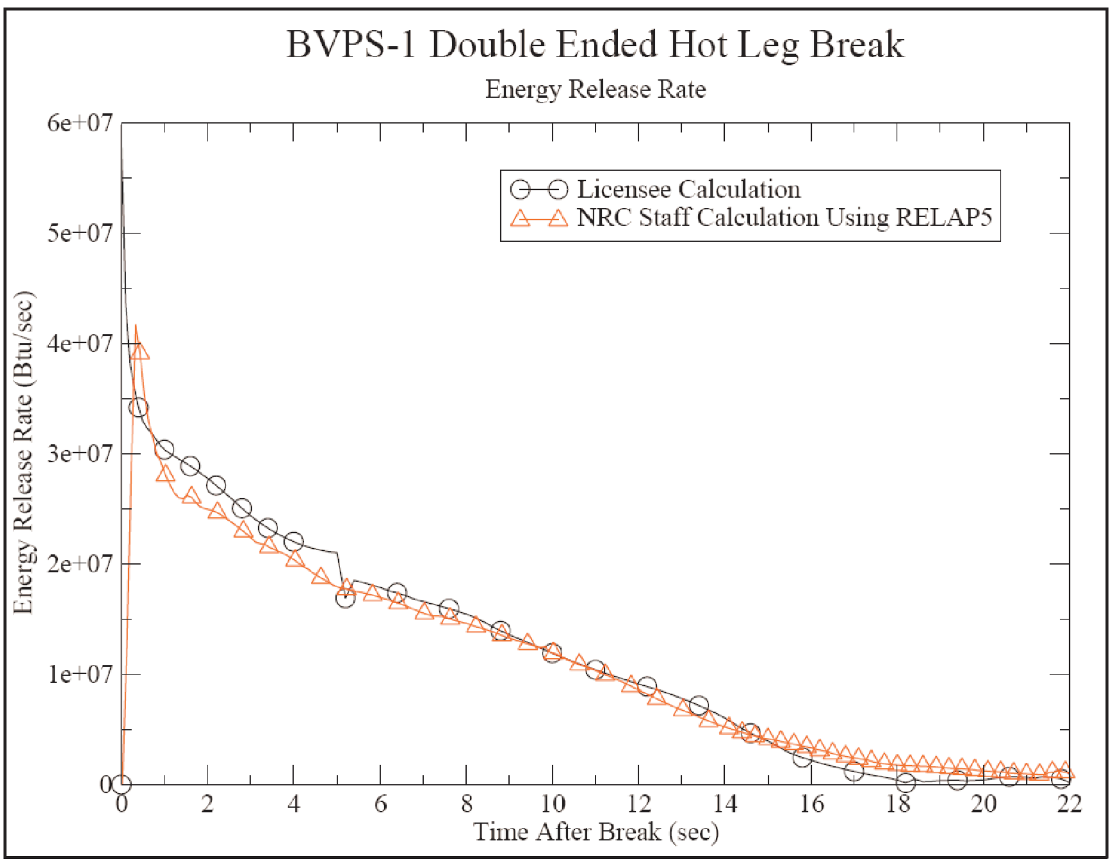

Figure 4: DEHL Break Energy Release Rate

Figure 5 shows the energy release rate as a function of time for the blowdown phase of the DEHL break compared to the licensee's calculation. The x-axis shows time after the break in seconds, the y-axis shows the sum of energy release rates from the vessel side and the pump

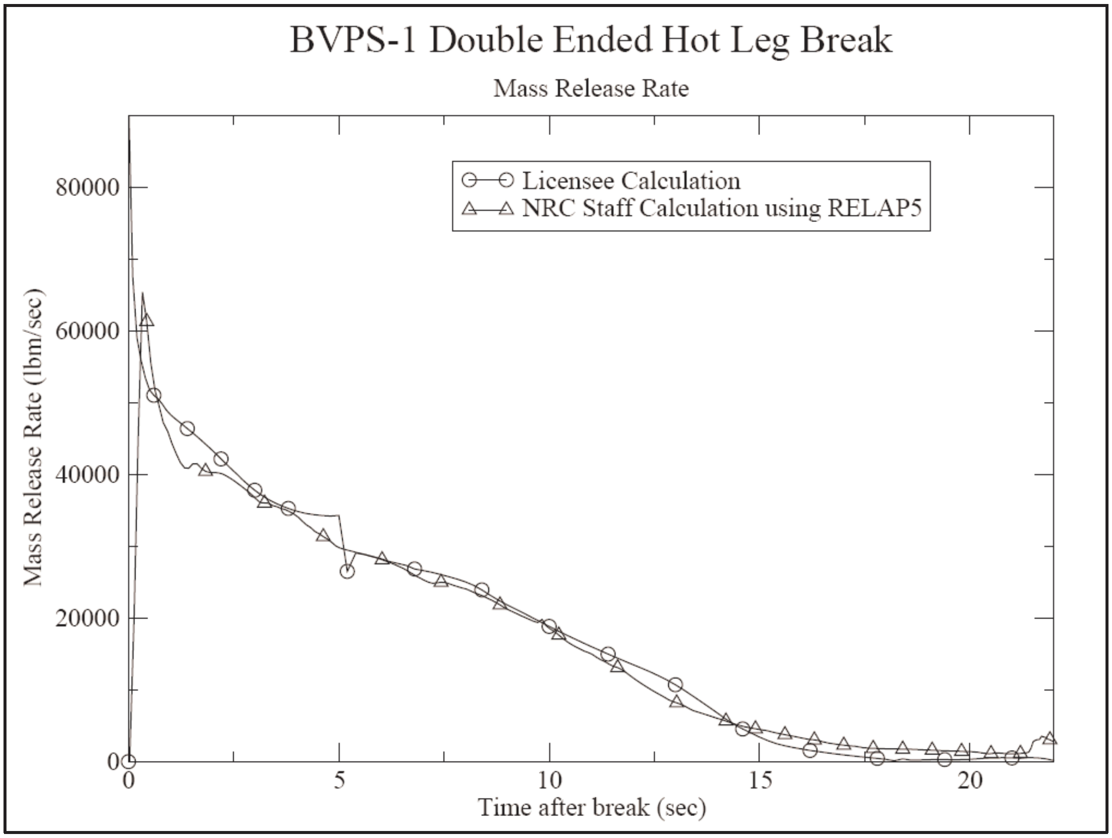

Figure 5: DEHL Break Mass Release Rate

side of the break. The licensee's and NRC staff's calculation of the energy release rate for the DEHL break show close agreement.

3.2 DEPS Break

The DEPS break combines the effects of the relatively high core-flooding rate, as in the hot-leg break, and the additional stored energy in the steam generators (SGs). As a result, the DEPS break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the reactor coolant system in calculating the releases to containment. The results of the NRC staff's mass and energy release calculation for the DEPS break are summarized in Figures 6 through 9. Figures 6 and 7 show the mass release rate as a function of time calculated by the NRC staff compared to the licensee's calculation. Figure 6 shows the blowdown in the first 25 seconds of the event. Figure 7 shows the long-term energy release rate up to 3600 seconds into the event. For both plots, the x-axis shows time after the break in seconds, the y-axis shows the sum of the mass release rates from the SG side and the pump side of the break in the suction leg.

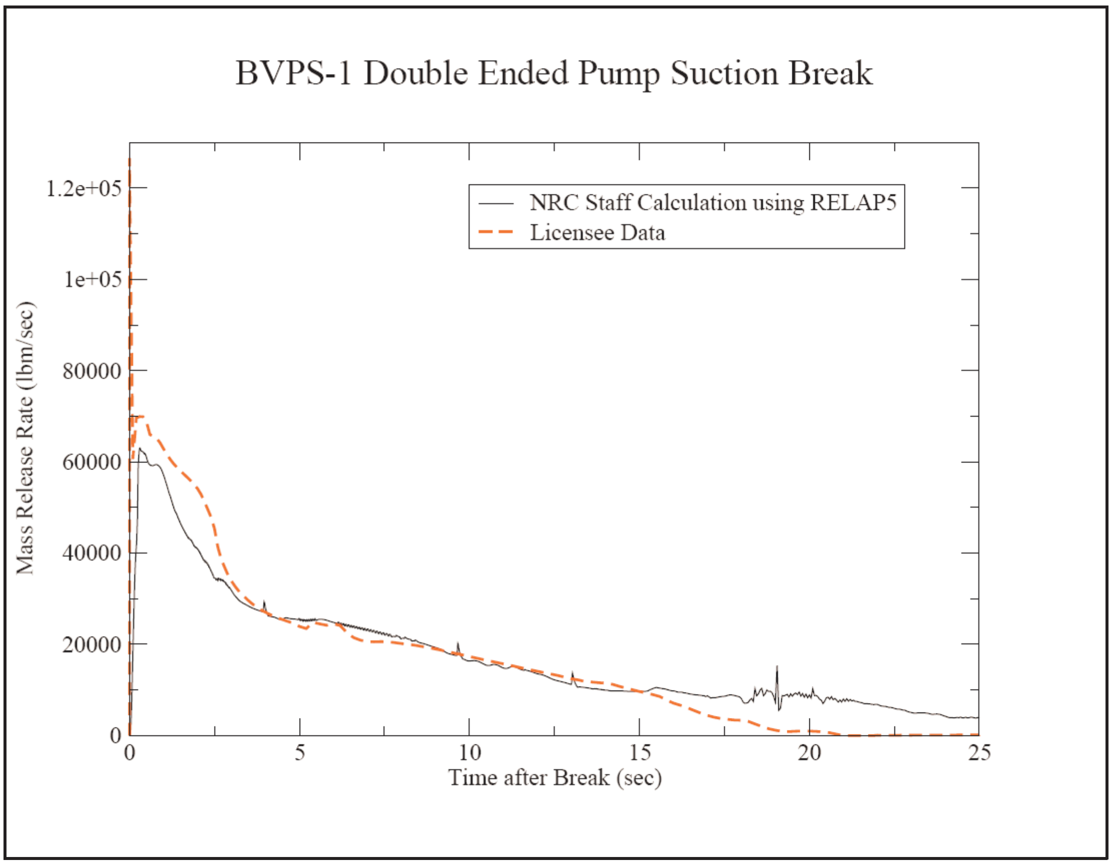

Figure 6: DEPS Break Mass Release Rate (0 to 25 seconds)

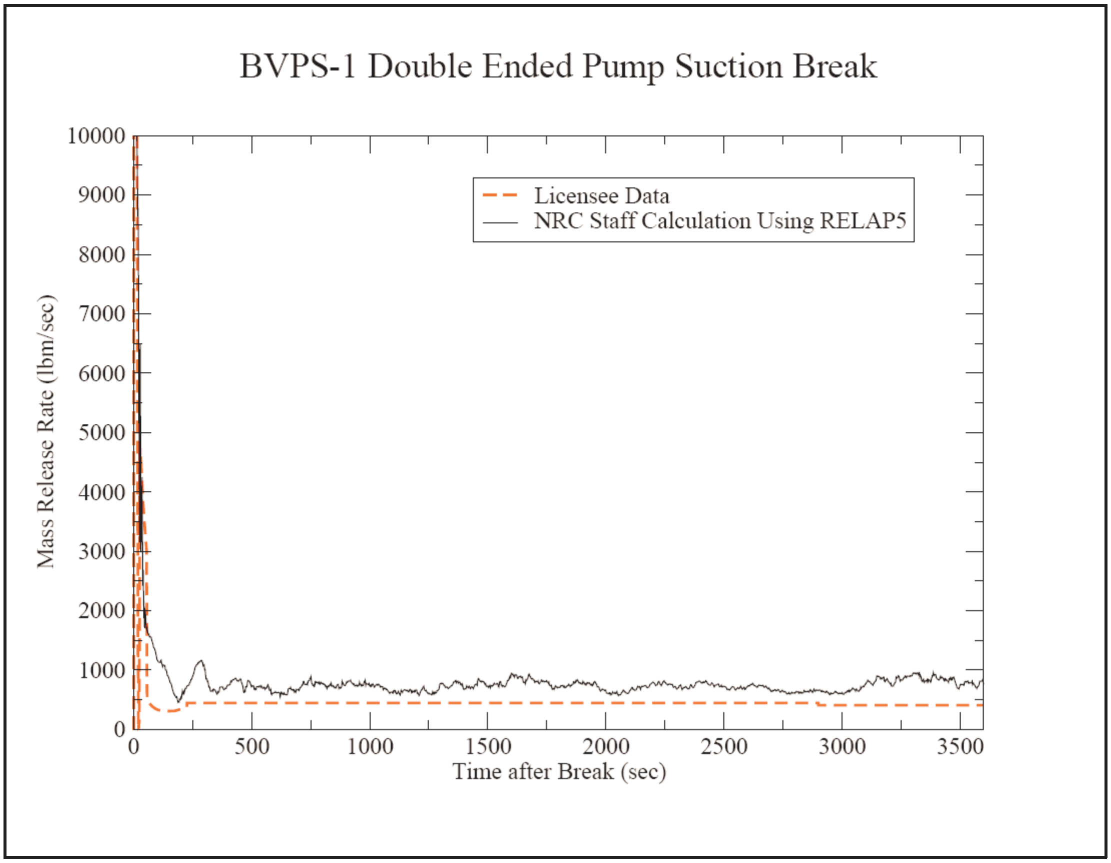

Figure 7: DEPS Break Mass Release Rate (0 to 3600 sec)

Figures 6 and 7 demonstrate that the mass release rate calculated by the licensee is in relatively close agreement with the NRC staff's calculation. In the long term, the mass release rate calculated by the NRC staff was higher than that of the licensee's calculation by about 50 percent. This is expected since the assumptions used by the NRC staff in the injection rate were also about 50 percent higher.

Figures 8 and 9 show the energy release rate as a function of time calculated by the NRC staff compared to the licensee's calculation. Figure 8 shows the blowdown in the first 25 seconds of the event. Figure 9 shows the longer term energy release rate up to 3600 seconds into the event. For both plots, the x-axis shows time after the break in seconds, the y-axis shows the sum of the energy release rates from the SG side and the pump side of the break in the suction leg.

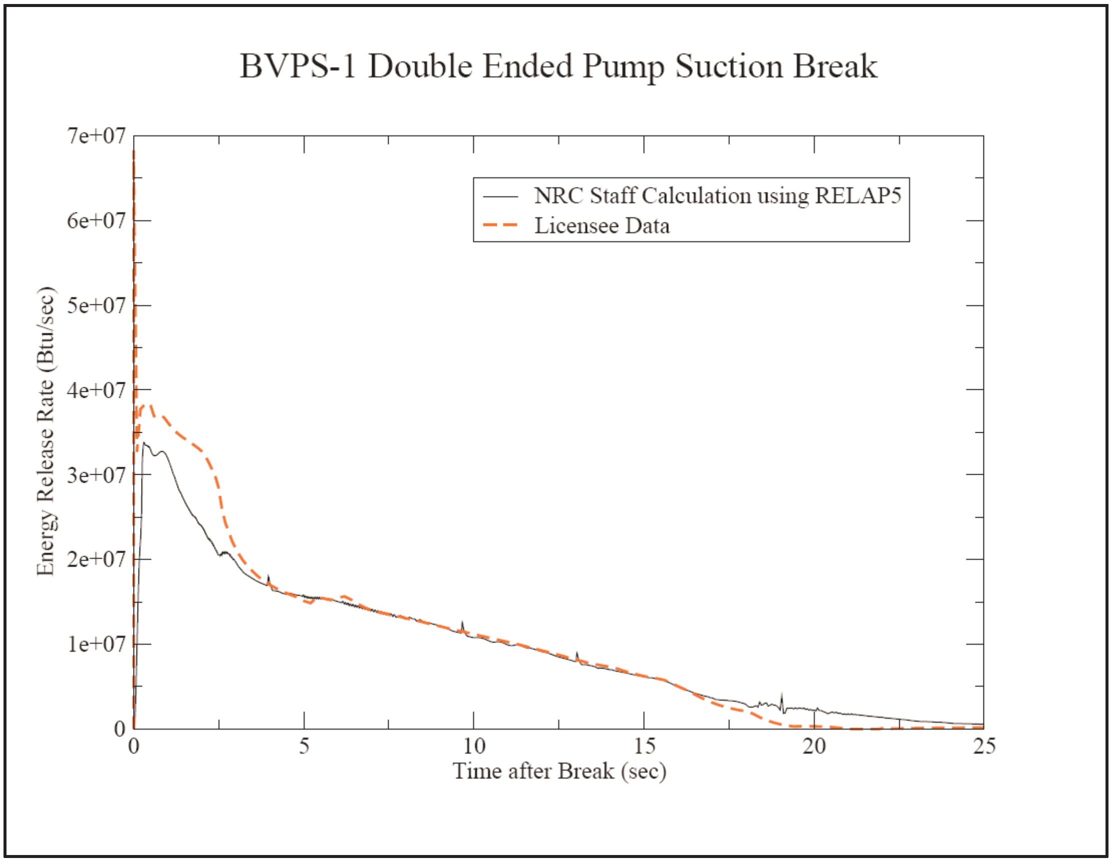

Figure 8: DEPS Break Energy Release (0 to 25 seconds)

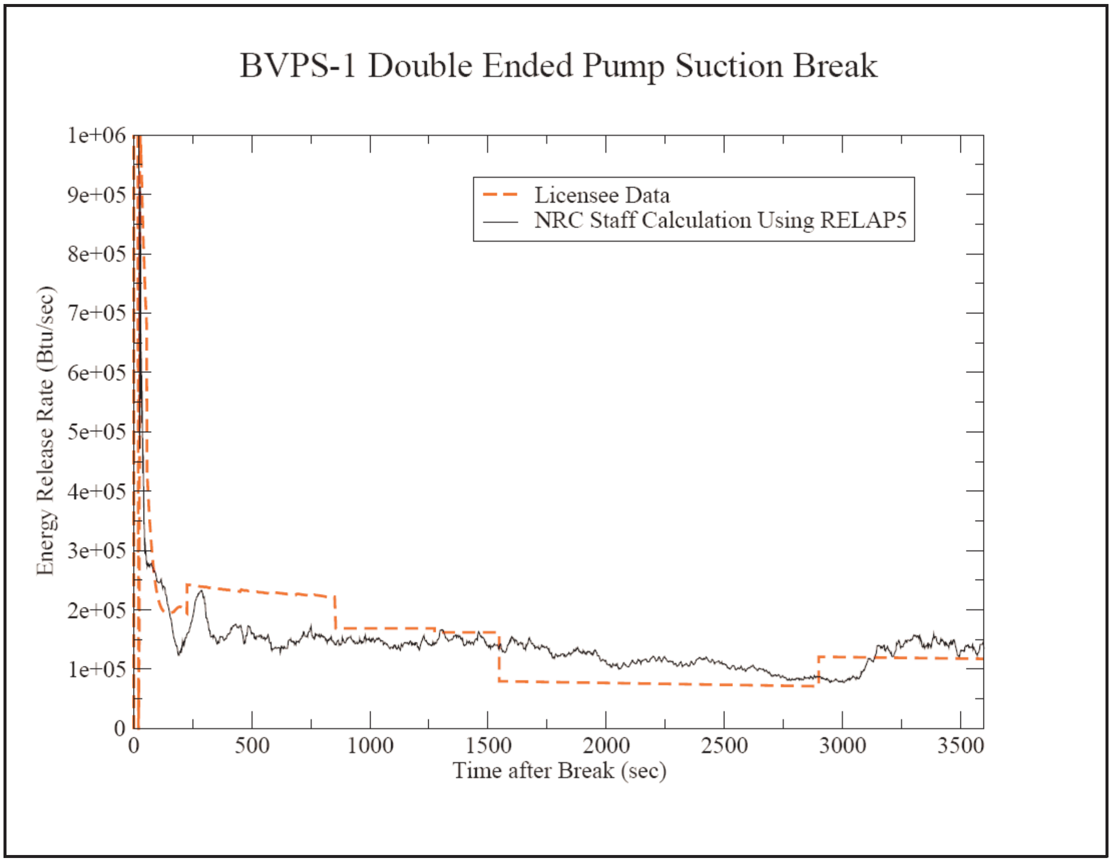

Figure 9: DEPS Break Energy Release (1 to 3600 seconds)

Figures 8 and 9 demonstrate that the energy release rate for the DEPS break calculated by the licensee agrees relatively well with the NRC staff's calculation despite the NRC staff using conservative decay heat and higher injection flow rate assumptions.

4.0 CONCLUSION

The licensee's calculations for mass and energy release rates are in agreement with the NRC staff's calculations using RELAP5 for both the DEHL and the DEPS loss-of-coolant accident events. The licensee's long-term mass release rate for the DEPS break is lower than the NRC staff's calculations, but the differences are not substantial and can be explained by the differences in the assumptions used in the NRC staff's calculations.