September 10, 2003

Mr. L. William Pearce
Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: SELECTIVE IMPLEMETATION OF ALTERNATE SOURCE TERM AND CONTROL ROOM HABITABILITY TECHNICAL SPECIFICATION CHANGES (TAC NOS. MB5303 AND MB5304)

Dear Mr. Pearce:

The Commission has issued the enclosed Amendment No. 257 to Facility Operating License No. DPR-66 and Amendment No. 139 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2). These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 5, 2002, as supplemented August 19 and December 2, 2002, and January 30, February 14, March 19 and 31, June 6 and 24, and September 5, 2003.

These amendments approve selective implementation of an alternative source term methodology for the loss-of-coolant accident (LOCA) and the control rod ejection accident (CREA), incorporation of ARCON96 methodology for release points associated with the LOCA and CREA, elimination of the control room emergency bottled air pressurization system, changes to the control room emergency ventilation system (CREVS), and a change to the BVPS-1 CREVS filter bypass leakage acceptance test criteria. The request to review related Updated Final Safety Analysis Report (UFSAR) pages submitted with your February 14, 2003, submittal was withdrawn in your March 31, 2003, submittal. The June 5, 2002, request to modify TS 3.6.2.2, "Containment Recirculation Spray System," by reducing the required recirculation spray heat exchanger minimum river/service water flow rate in surveillance requirement (SR) 4.6.2.2.e.3, was withdrawn by your March 19, 2003, letter. The changes were no longer necessary as Amendment Nos. 252 and 132 for BVPS-1 and BVPS-2, respectively, relocated SR 4.6.2.2.e.3 to each unit's UFSAR. The changes related to conversion of the BVPS-1 and 2 containments from subatmospheric to atmospheric operating conditions was withdrawn by your letter dated September 5, 2003. As discussed with and agreed to by your staff, the enclosed amendments include a 60-day implementation period vice the 120-day implementation period requested in your March 31, 2003, submittal, as the full 120-day period is no longer needed.

A copy of our safety evaluation is also enclosed. The enclosed Notice of Partial Withdrawal has been filed with the Office of the *Federal Register* for publication. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures: 1. Amendment No. 257 to DPR-66
2. Amendment No. 139 to NPF-73
3. Safety Evaluation
4. Notice of Partial Withdrawal

cc w/encls: See next page

A copy of our safety evaluation is also enclosed. The enclosed Notice of Partial Withdrawal has been filed with the Office of the *Federal Register* for publication. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Amendment No. 257 to DPR-66
2. Amendment No. 139 to NPF-73
3. Safety Evaluation
4. Notice of Partial Withdrawal

cc w/encls: See next page

DISTRIBUTION:

PUBLIC	MO'Brien	ACRS	PDI-1 R/F	TColburn	BPlatchek, RGN-I
OGC	MTschiltz	RLaufer	GHill (4)	RDennig	SLaVie
EForrest	ADrozd	FReinhart	LRaghavan		

ADAMS Accession Number: ML032530204
Incoming: ML , Package: ML
*See previous concurrence

OFFICE	PDI-1/PM	PDI-1/LA	SPSB/SC*	PDI-1/SC	OGC*
NAME	TColburn	MO'Brien	RDennig	RLaufer	LZacarri
DATE	9/10/03	9/10/03	8/26/03	9/10/03	8/29/03

OFFICIAL RECORD COPY

cc:

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

FirstEnergy Nuclear Operating Company
Regulatory Affairs/Performance Improvement
Larry R. Freeland, Manager
Beaver Valley Power Station
Post Office Box 4, BV-A
Shippingport, PA 15077

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

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

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

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

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

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

Rich Janati, Chief
Division of Nuclear Safety
Bureau of Radiation Protection
Department of Environmental Protection
Rachel Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469

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

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

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

FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
ATTN: M. P. Pearson, Director
Services and Projects (BV-IPAB)
Post Office Box 4
Shippingport, PA 15077

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

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 257
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 5, 2002, as supplemented August 19 and December 2, 2002, and January 30, February 14, March 19 and 31, June 6 and 24, and September 5, 2003, 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. 257, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 10, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 257

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 3-36	3/4 3-36
3/4 7-16	3/4 7-16
3/4 7-16a	3/4 7-16a
3/4 7-17	3/4 7-17
3/4 7-18	3/4 7-18
3/4 7-18a	3/4 7-18a
3/4 7-18b	3/4 7-18b

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
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 5, 2002, as supplemented August 19 and December 2, 2002, and January 30, February 14, March 19 and 31, June 6 and 24, and September 5, 2003, 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. 139, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 10, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 139

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 7-15	3/4 7-15
3/4 7-16	3/4 7-16
3/4 7-17	3/4 7-17
3/4 7-17a	3/4 7-17a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 257 AND 139 TO FACILITY OPERATING

LICENSE NOS. DPR-66 AND NPF-73

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By application dated June 5, 2002, as supplemented by letters dated August 19, and December 2, 2002, and January 30, February 14, March 19 and 31, June 6 and 24, and September 5, 2003, the FirstEnergy Nuclear Operating Company (FENOC, et al., the licensee), requested changes to the Technical Specifications (TSs) for Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2). The supplements dated August 19 and December 2, 2002, and January 30, February 14, March 19 and 31, June 6 and 24, and September 5, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed except as noted below, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 10, 2002, (67 FR 75876). The February 14, 2003, submittal requested the scope of the review be expanded by including in the scope of the review related Updated Final Safety Analysis Report (UFSAR) page changes, but this request was withdrawn in the March 31, 2003, submittal. Additionally, a portion of the requested review was withdrawn in the March 19, 2003, submittal, as these changes were no longer necessary. The portion of the proposed application related to conversion of the BVPS-1 and 2 containments from subatmospheric to atmospheric operating conditions was withdrawn by letter dated September 5, 2003.

The proposed changes would revise the TSs to allow approval of selective implementation of an alternative source term (AST) methodology for the loss-of-coolant accident (LOCA) and the control rod ejection accident (CREA), incorporation of ARCON96 methodology for release points associated with the LOCA and CREA, elimination of the control room emergency bottled air pressurization system (CREBAPS), changes to the control room emergency ventilation system (CREVS), and a change to the BVPS-1 CREVS filter bypass leakage acceptance test criteria.

In support of these changes, FENOC evaluated the impact of the changes on the previously analyzed radiological consequences of postulated LOCA and CREA design-basis accidents (DBAs) utilizing a selective implementation of an alternative source term (AST). These analyses were performed assuming a licensed core power of 2900 megawatts thermal (MWt) plus 0.6 percent uncertainty (2918 MWt). The current authorized power level for BVPS-1 and 2 is 2689 MWt.

The proposed control room habitability changes also impact the results of the main steam line break (MSLB) accident and, for BVPS-1 only, the locked rotor accident (LRA). These latter analyses were performed using the current licensing basis source term, dose acceptance criteria, and the current licensed core power. The remaining DBAs are not affected by the proposed change to the control room habitability systems.

1.1 Selectively Implement an AST

FENOC proposes to modify the BVPS-1 and 2 design basis for the LOCA and CREA radiological analyses to replace the current accident source term with an AST and to replace the previous whole body and thyroid accident dose guidelines with the total effective dose equivalent (TEDE) criteria of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.67(b)(2). FENOC has proposed a selective implementation of the AST and TEDE criteria, as described in Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” These analyses were performed assuming a reactor power of 2918 MWt (plus uncertainty), a value greater than that currently authorized for BVPS-1 and 2. (Approval of this amendment does not provide authorization to operate at this uprated power level). FENOC had previously received approval for a selective implementation of the AST for the fuel handling accident (FHA) at BVPS-1 and 2, on August 30, 2001, with Amendment Nos. 241 and 121, respectively.

1.2 Change TS 3.7.7.1, “Control Room Emergency Habitability Systems” (BVPS-1) and

1.3 Change TS 3.7.7, “Control Room Emergency Air Cleanup And Pressurization System” (BVPS-2)

FENOC proposes to remove the operability requirements for the control room bottled air pressurization system, including Limiting Condition for Operation (LCO) “b” and associated action statements and surveillance requirements. FENOC also proposes to revise surveillance requirements (SRs) 4.7.7.1.1.c.1 and 4.7.7.1.2.c.1 to change the emergency ventilation filtration system bypass leakage testing requirements from <1 percent to <0.05 percent for the high efficiency air particulate (HEPA) and charcoal filters in the BVPS-1 TSs, to be consistent with the BVPS-2 TS requirement.

1.4 Change TS Table 4.3-3, “Radiation Monitoring Instrumentation Surveillance Requirements” (BVPS-1 only)

FENOC proposes, for radiation monitors RM-RM-218 A & B, to remove reference to the control room emergency bottled air pressurization system (CREBAPS) for the channel function test. This change is consistent with the proposed de-activation of CREBAPS. Note that the operability requirements for these two monitor channels for indication and alarm remain effective.

2.0 REGULATORY EVALUATION

This safety evaluation input addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results.

FENOC determined that the DBA LOCA, CREA, and MSLB analyses for both units would be affected by the proposed changes. In addition, the LRA at BVPS-1 is affected. In assessing the impact of the proposed changes on the DBA LOCA and CREA, FENOC opted to use an AST. The use of an AST is addressed in 10 CFR 50.67, “Accident Source Term,” which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses. Regulatory guidance for the implementation of these ASTs is provided in RG 1.183. A licensee seeking to use an AST is required, pursuant to 10 CFR 50.67 to apply for a license amendment. An evaluation of the consequences of affected DBAs is required to be included with the submittal. FENOC’s application of June 5, 2002, as supplemented by letter dated January 30, 2003, addresses these requirements in proposing to selectively use the AST described in RG 1.183 as the DBA source term in the evaluation of the radiological consequences of a DBA LOCA and CREA at BVPS-1 and 2. As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion 19 (GDC 19), “Control Room,” as the BVPS-1 and 2 licensing basis for LOCA and CREA events. The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183, and GDC 19. Except where FENOC proposed a suitable alternative, the U.S. Nuclear Regulatory Commission (NRC) staff utilized the regulatory guidance in RG 1.183, and the Standard Review Plan (SRP), Section 15.0-1, “Radiological Consequence Analyses Using Alternative Source Terms,” in performing this review. The NRC staff also considered relevant information in the BVPS-1 and 2 UFSARs and the TSs.

For the MSLB and LRA analyses, the regulatory requirements for which the NRC staff based its acceptance are the accident dose guidelines in 10 CFR 100.11, as supplemented by accident-specific criteria in Section 15 of the SRP, and GDC 19, “Control room,” as supplemented by Section 6.4 of the SRP. Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance provided in the following documents in performing this review. FENOC utilized newly calculated control room atmospheric relative concentrations (χ/Q) values in performing these assessments.

- SRP Section 15.1.5, Appendix A, “Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR”
- SRP Section 15.3.3, “Reactor Coolant Pump Rotor Seizure”
- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” June 2003 (issued for public comment as draft in December 2001)

The NRC staff also considered relevant information in the BVPS-1 and 2 UFSARs and TSs in assessing the adequacy of the licensee’s dose assessment analyses of the radiological consequences of the proposed changes.

With respect to the other control room habitability changes, specifically, the elimination of the CREBAPS, the following regulatory criteria apply.

Section 50.36 provides requirements for the inclusion of an LCO in the TSs based on the following four criteria:

Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Section 50.36(c)(3) addresses the need for surveillance requirements. Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

Appendix A to 10 CFR Part 50, GDC 19, “Control room,” contains requirements for maintaining a habitable environment in the control room under normal and accident conditions, including LOCAs.

SRP, Section 9.4.1, “Control Room Area Ventilation Systems,” and SRP Section 6.4 provide guidance for the review of toxic gas and pressurization systems.

RG 1.52, Revision 3, provides guidance describing the design, inspection and testing criteria for air filtration and adsorption units of post-accident engineered safety feature atmosphere cleanup systems.

The NRC staff considered the above regulations, design criteria and guidance in evaluating the licensee’s proposed changes with respect to control room habitability.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the regulatory and technical analyses performed by FENOC in support of its proposed license amendment. The NRC staff limited its review to the radiological consequences of DBAs. Information regarding these analyses was provided in Section 4 of Enclosure 1 and Section 5 of Enclosure 2 of FENOC’s June 5, 2002, submittal, and in a supplement dated January 30, 2003. The NRC staff also had the benefit of teleconferences with representatives of FENOC on October 8, 2002, and November 7, 2002, and a public meeting on December 11, 2002. The NRC staff reviewed the assumptions, inputs, and methods used by FENOC to assess these impacts. The assumptions found acceptable are tabulated in Table 2. Although the NRC staff performed independent calculations to confirm the conservatism of many of the FENOC analyses, the findings of this safety evaluation input are based on the descriptions of the FENOC analyses and other supporting information docketed by FENOC. Only docketed information was relied upon in making this safety finding.

FENOC performed an evaluation of the impact of the proposed changes on the previously analyzed DBAs addressed in the BVPS-1 and 2 UFSARs. The DBAs were dispositioned as follows:

- LOCA. Re-analyzed using the AST and an assumed reactor power level of 2918 MWt. Single analysis addresses both units. Exclusion area boundary (EAB), low population zone (LPZ), control room, and emergency response facility (ERF) doses were calculated.

- CREA. Re-analyzed using the AST and an assumed reactor power level of 2918 MWt. Single analysis addresses both units. EAB, LPZ, and control room doses were calculated.

- MSLB. These analyses were updated to reflect the CREBAPS changes. Separate analyses were done for each unit. EAB, LPZ, and control room doses were calculated.

- LRA. The BVPS-1 analysis is affected by the CREBAPS change and was re-evaluated by the licensee using the TID14844 source term and the current reactor power level plus uncertainty. Since the current BVPS-2 analysis took no credit for control room isolation or pressurization, the current BVPS-2 analysis results remain bounding. EAB, LPZ, and control room doses were calculated.

Since none of these remaining DBA analyses credited control room isolation and pressurization, there are no impacts from the CREBAPS changes and the analyses results remain bounding.

3.1 Core Inventory

The inventory of fission products in the reactor core is based on a reactor power level of 2918 MWt. This core inventory supports licensed thermal powers of up to 2900 MWt. The current licensed thermal power for both BVPS units is 2689 MWt. The higher power level was used in determining the core inventory so that the assessment would be applicable to future uprated conditions as well as the current licensed power. The core inventory was determined using the NRC-sponsored SCALE computer code package and input parameters that are bounding for both BVPS units.

3.2 Atmospheric Dispersion

FENOC re-calculated the atmospheric relative concentrations (χ/Q) for the control room intakes in support of this amendment request. Numerous combinations of release points and control room outside air intakes were considered. The combinations considered and the resulting χ/Q values were tabulated in Tables 5.3.4-2 and 5.3.4-3 of the June 5, 2002, submittal. Tables 2.0.5-1 and 2.0.5-2 of the January 30, 2003, letter provide tables of input parameters used in the ARCON96 code runs. These χ/Q values were calculated using the NRC-sponsored ARCON96 computer code as described in NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wakes.” Meteorological data collected on the BVPS-1 and 2 meteorological tower between January 1, 1990, and December 31, 1994 (i.e., 5 years) were used in this assessment. All releases were treated conservatively as ground level releases. The containment surface cases were treated as ground level releases with the containment treated as a diffuse source. The NRC staff performed quality checking on the submitted data to assess its validity and representativeness, comparing data distributions to corresponding tabulations in the BVPS-1 and 2 UFSARs. The NRC staff performed independent calculations of selected release point and receptor combinations.

Based on its review of the information provided by the licensee, the NRC staff finds that FENOC used analysis methods and assumptions acceptable to the NRC staff in determining the χ/Q values. With regard to the control room values, the NRC staff reviewed the use of the methodology and found it to be consistent with current NRC staff positions on the use of ARCON96. The NRC staff performed a qualitative review of the code results and deemed them to be reasonable. The staff finds the revised χ/Q values acceptable. The NRC staff did not review the offsite χ/Q values for this amendment request as these values had been previously reviewed by the NRC staff.

3.3 Control Room Habitability

Currently the BVPS control room, which is common to both units, operates in the zone isolation with bottled air pressurization mode as defined in SRP Chapter 6.4. However, control room isolation is not credited in the analyses for several DBAs. In the current configuration, receipt of a containment isolation phase B (CIB) signal, a high control room radiation alarm, or a manual actuation for either unit will isolate the outside air intakes, stop the normal make-up intake fans, actuate the control room emergency breathing air pressurization system (CREBAPS), and start a 60-minute timer. At 60 minutes following the actuation signal (N.B., prior to depletion of the air bottles), filtered intake fans of the control room emergency ventilation system (CREVS) will start. After the proposed modifications are implemented, CREBAPS will be disabled and CREVS will start, without planned delay, following CIB or manual actuation. The radiation monitor actuation will no longer be credited. In this proposed configuration, the BVPS control room will operate in the zone isolation with filtered pressurization mode.

FENOC has performed tracer gas measurements of the unfiltered inleakage to the control room in both the isolated mode and pressurized modes. An unfiltered inleakage of 10 cfm due to ingress and access was added to the mean values for the tracer gas measurements to arrive at the unfiltered inleakage values assumed in the dose calculations. For the isolation mode, the unfiltered inleakage is assumed to be 300 cfm; for the pressurized mode, 30 cfm. CREVS intake filters are assumed to be 99 percent efficient for particulates and 98 percent efficient for elemental and organic iodine species. These efficiencies reflect the more limiting HEPA and charcoal filter test criteria proposed for BVPS-1 as part of this amendment (applied to BVPS-2 in a previous amendment).

At the onset of any event, the control room ventilation system is operating with normal make-up intake flow of 500 cfm. As noted above, receipt of an actuation signal would actuate control room isolation and actuate the CREVS. However, a loss of offsite power is assumed to occur coincident with many DBAs, removing power from intake fans and intake isolation dampers. Power is restored by an emergency diesel generator in 77 seconds. With power restored, the CREVS realign, and CREVS start. For analysis purposes, FENOC assumes that the first CREVS fan fails to start. Given circuitry delays, FENOC expects a 137-second delay on the automatic starting of CREVS on receipt of an actuation signal. Nonetheless, FENOC conservatively assumes in dose calculations that CREVS is actuated by operator manual actions at T=30 minutes. Once CREVS starts, the filtered intake flow rate is expected to vary between 600 and 1030 cfm. Sensitivity analyses by FENOC have shown that the lower flow rate is generally more limiting since the higher flow rate results in a greater dilution of control room atmosphere concentrations. Accident-specific differences in the modeling of control room operator doses will be addressed in the following accident discussions.

3.4 ERF Habitability

BVPS-1 and 2 is served by a single ERF that supports both units. During normal plant operation, the ERF ventilation system intake flow rate of 3800 cfm is processed through HEPA filters. Unfiltered inleakage during normal operation is estimated at 2090 cfm. Following a LOCA, the ERF is manually isolated and the ventilation system is switched to emergency filtered recirculation mode at T=30 minutes. The recirculation flow rate is 3800 cfm and the unfiltered inleakage is estimated to be 910 cfm.

For the purpose of analysis, FENOC assumed that there is no ERF structure, that is, the personnel in the ERF are modeled as unsheltered and unshielded point receptors in the environment. In the interest of assessing the impact of a build-up of radiogases in the ERF that could create radiation exposures after the plume had cleared the area, the NRC staff performed its confirmatory analyses modeling the ERF as a structure with intake and removal processes.

3.5 LOCA

A DBA LOCA is a failure of the reactor coolant system (RCS) that results in the loss of reactor coolant which, if not mitigated, could result in fuel damage including core melt. Analyses are performed using a spectrum of RCS break sizes to evaluate fuel and emergency core cooling system (ECCS) performance. A large break LOCA is postulated as the failure of the largest pipe in the RCS. RG 1.183 establishes the large-break LOCA as the licensing basis LOCA with regards to radiological consequences since this represents the larger challenge to plant safety features designed to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective. Evaluation of the effectiveness of plant safety features, such as ECCS, has shown that core melt is unlikely. The objective of this DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective in preventing core damage.

3.5.1 Core Fission Product Release

During a DBA LOCA, it is assumed that the initial fission product release to the containment will last 30 seconds and will involve all of the radioactive materials dissolved or suspended in the RCS liquid. The gap release phase begins with the onset of fuel cladding failure and is assumed to continue to 30 minutes. As the core continues to degrade, the gap release phase ends and the early in-vessel release phase commences. This phase continues for 1.3 hours. The inventory in each release phase is assumed to be released at a constant rate over the duration of the phase and starting at the onset of the phase. The LOCA source term release fraction, timing characteristics, and radionuclide grouping are tabulated in Table 1.

Table 1

Radionuclide Group	Gap Release Phase *(0.5 Hours)*	Early In-Vessel Phase *(1.3 Hours)*
Noble Gases *(Xe, Kr, Rn, H)*	0.05	0.95
Halogens *(I, Br)*	0.05	0.35
Alkaline Metals *(Cs, Rb)*	0.05	0.25
Tellurium Group *(Te, Sb, Se)*	0	0.05
Barium *(Ba, Sr)*	0	0.02
Noble Metals *(Ru, Rh, Pd, Mo, Tc, Co)*	0	0.0025
Cerium Group *(Ce, Pu, Np)*	0	0.0005
Lanthanides *(La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)*	0	0.0002

Fission products are released from the core into the containment and are assumed to mix instantaneously and homogeneously throughout the free volume of the containment. The release into the containment is assumed to terminate at the end of the early in-vessel phase. FENOC assumes that the iodine released to the containment is comprised of 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic forms. This iodine speciation is appropriate if the containment sump pH is maintained at a value of 7.0 or higher. This is accomplished at BVPS-1 and 2 by chemical injection into the quench spray system. Once dispersed in the containment, the release to the environment is assumed to occur through four pathways:

- Release from containment vacuum system prior to isolation
- Release from containment leakage
- Sump water leakage from ECCS systems outside of the containment
- Release from refueling water storage tank (RWST) due to ECCS backleakage

3.5.2 Containment Vacuum System Release

FENOC assumes that the containment vacuum system is operating at the start of the LOCA providing a path for release to the environment. This line is projected to be isolated prior to T=5 seconds as a result of a containment isolation signal. Since the onset of fission product releases from the fuel occurs at 30 seconds, this pathway is isolated prior to fuel damage occurring. For purposes of analysis, FENOC assumes that the entire RCS inventory of volatile radionuclides is released to the containment, from where it enters the environment at a rate of 2200 cfm for 5 seconds. Since fuel damage and containment sprays will not have commenced in this period, FENOC assumes that the chemical form of the iodine released from the RCS is 97 percent elemental and 3 percent organic. The NRC staff finds these assumptions to be acceptable.

3.5.3 Containment Leakage Release

The containment building holds up the majority of the radioactivity released from the core. During a LOCA, FENOC assumes that the containment leaks at a rate of 0.1-percent volume per day for the first 24 hours and 0.05-percent volume per day for days 2 through 30. These leakage assumptions differ from the current licensing basis which assumed the cessation of containment leakage at about one hour following the LOCA. These leakage assumptions were made in support of a proposed containment conversion from subatmospheric to atmospheric operating conditions for re-analysis of containment pressure transient analyses (the proposed containment conversion has not been approved). However, this assumed leakage exceeds that currently provided for by the design basis (i.e., 0.1-percent volume per day for 1 hour). This is a conservative situation. FENOC also conservatively assumes no collection or filtration of this containment leakage.

The containment atmosphere at BVPS-1 and 2 is sprayed by the containment spray systems during a LOCA. These systems are automatically started by containment pressure instrumentation. There are two systems, the quench spray system (QSS) and the recirculation spray system (RSS). The QSS would actuate prior to about T=85 seconds and is available until depletion of the RWST inventory, at about T=720 seconds. At this time, the RSS actuates and provides containment spray for about 4 days. The containment spray is effective at removing particulate aerosols and elemental iodine from the containment atmosphere. Conservatively, FENOC does not credit iodine removal by the containment sprays prior to T=722 seconds or after 4 days. Because of the configuration of equipment and internal structures within the containment the sprays do not cover the full containment volume. FENOC modeled the containment as being comprised of two regions–sprayed, and unsprayed. Based on evaluations of spray nozzle coverage and containment arrangement, FENOC projects that 63 percent of the containment free volume is sprayed. FENOC assumes that the containment atmosphere is mixed following a LOCA by four mechanisms, justifying an assumed mixing rate between the sprayed and unsprayed regions of 2 hr^{-1}, as provided for in the SRP. FENOC did not quantify the effectiveness of these mechanisms, but identified them as qualitative justification for their conclusion that adequate containment mixing would occur to support the default SRP assumption that the mixing rate between the sprayed and unsprayed regions would be two unsprayed region volumes per hour. The NRC staff concurs.

FENOC assumed that containment sprays were effective for particulates and elemental iodine. No credit for spray removal was assumed for noble gases or for organic forms of iodine. The effectiveness of the sprays for fission product scrubbing is represented by the removal rate (often referred to as spray coefficients or spray lambda, λ). Although there are several aerosol phenomena that promote the depletion of aerosols from the containment, the FENOC particulate removal calculation only takes credit for diffusiophoresis and the removal effectiveness of the sprays. Agglomeration was considered. FENOC asserts that if the natural removal phenomena were included, the total removal effectiveness would increase even though the effectiveness of the spray removal would be slightly reduced. Only gravitational settling of aerosols is credited for the unsprayed region. FENOC assumed a mass mean spray droplet radius of 350 microns for the time period of 722 to18,000 seconds and a mass mean spray droplet radius of 518.5 microns for the time period of 18,000 to 346,000 seconds. In practice, a large distribution of spray droplet sizes would result, most of which having radii less than that assumed. Since the efficiency of spray removal increases with decreasing spray droplet radii, the single droplet size is considered to be conservative. Although a single spray droplet size was used, a distribution of aerosol sizes was considered and the spray removal efficiency was determined for each aerosol size bin.

FENOC presented the calculated removal rate of particulates by sprays and diffusiophoresis versus time after the accident by means of a graph (Figure 5.3.6-1 of the June 5, 2002, submittal). Values range from about 6.0 hr^{-1} at 722 seconds post-accident, to a maximum value of about 57 hr^{-1} at about 5200 seconds, and tapering off to about 2 hr^{-1} at 12600 seconds. Similarly, the removal rate of particulates within the unsprayed region by gravitational settling versus time after the accident is shown by means of a graph (Figure 5.3.6-2 of the June 5, 2002, submital). Values range from about 0.005 hr^{-1} at 722 seconds post-accident, to a maximum value of about 0.95 hr^{-1} at about 10200 seconds, and tapering off to about 0.89 hr^{-1} at 18000 seconds.

These particulate removal rates were determined using the Stone and Webster proprietary computer program SWNAUA as a proposed alternative to the methods identified in SRP Section 6.5.2 and RG 1.183. SWNAUA is described as a proprietary variant of the NAUA aerosol deposition in containment code that was modified for incorporation in the Source Term Code Package, a code package for assessing severe accidents. Although the NRC staff has previously accepted removal rates determined using SWNAUA for use in design-basis analyses, the NRC staff has not endorsed the use of SWNAUA for general use in design-basis analyses. A topical report that provides a technical derivation of the SWNAUA modeling and a description of its verification and validation has not been docketed for NRC staff review and endorsement. The NRC staff's consideration of the use of SWNAUA in this plant-specific application does not change the unreviewed status of SWNAUA. In its review of the proposed spray removal rates, the NRC staff reviewed the brief overview of the code model provided by FENOC, reviewed the analysis inputs and assumptions, evaluated the reasonableness of the estimated removal rates, and reviewed the use of the estimated removal rates in the radiological analyses. The NRC staff's acceptance of the removal rates proposed by FENOC does not constitute endorsement of the use of the SWNAUA Code as an approved analysis methodology for design-basis analyses at BVPS-1 and 2 or at any other facility. Subject to this clarification, the NRC staff finds that FENOC's modeling of containment spray removal is acceptable as are the removal rates shown in Figures 5.3.6-1 and 5.3.6-2 of the June 5, 2002, submittal.

FENOC utilized the LOCTIC Code to develop the input parameters (e.g., pressure, temperature, and steam condensation rates) used with the SWNAUA Code. LOCTIC is the current BVPS-1 and 2 licensing basis methodology for evaluating the containment transient response. FENOC has proposed the use of the MAAP5 Code as a replacement for LOCTIC in evaluating the containment response. However, FENOC has stated that LOCTIC will remain the licensing basis code for developing inputs to the radiological analyses. LOCTIC calculates lower steam condensate rates than does MAAP5, a conservative situation with regard to fission product removal. The NRC staff concurs with the use of LOCTIC for this purpose. This safety evaluation does not evaluate the use of MAAP5 for evaluating the containment response.

In the June 5, 2002, submittal, FENOC stated the assumption that the elemental iodine plates out onto the particulate form and that the spray removal rate for the elemental iodine would be assumed to be equal to that determined for the particulate spray removal, but less than the maximum value of 20 hr^{-1} permitted by SRP 6.5.2. This mechanism is not supported by any empirical data and is not acceptable to the NRC staff. In discussions with the NRC staff, FENOC clarified this statement as a simplifying assumption to explain their conservative use of the particulate removal rates for the elemental iodine removal. As noted above, the NRC staff agrees that the use of the particulate removal rates for elemental iodine is conservative, but not on the basis of the unsupported statement. Using the model of SRP 6.5.2, a minimum elemental iodine plateout coefficient of 2 hr^{-1} was determined. No credit was taken for elemental iodine removal in the unsprayed region. Consistent with the guidance of SRP 6.5.2, FENOC established a maximum decontamination factor (DF) of 200 for elemental iodine.

3.5.4 Sump Water Leakage from ECCS Systems Outside of the Containment

During a DBA LOCA, some fission products released from the fuel will be carried to the containment sump via spillage from the RCS or by transport of activity from the containment atmosphere to the sump by containment sprays and natural processes such as deposition and plateout. During the initial phases of a LOCA, safety injection and containment spray systems draw water from the RWST. At about 5 minutes after the start of the event, these systems start to draw water from the containment sump instead. This recirculation flow causes contaminated water to be circulated through piping and components outside of the containment where small amounts of system leakage could provide a path for the release of fission products to the environment.

FENOC assumes that all of the radioiodines released from the fuel are instantaneously moved to the containment sump. Noble gases are assumed to remain in the containment atmosphere. The remaining radionuclides in Table 1 above are aerosols or particulates that will not become airborne on release from the ECCS. This source term assumption is conservative in that all of the radioiodines released from the fuel are credited in both the containment atmosphere and containment sump. FENOC assumes that the leakage rate is two times the expected value, or 11400 cc/hour. Since the temperature of this fluid is less than 212 °F, FENOC assumes that 10 percent of the entrained iodine activity is released to the atmosphere of the surrounding auxiliary building. FENOC assumes that this activity is exhausted without holdup, mixing, or filtration, and that the chemical form of the iodine released is 97 percent elemental and 3 percent organic.

3.5.5 Release From RWST due to ECCS Backleakage

Although the RWST is isolated during recirculation, design leakage through ECCS valving provides a pathway for backleakage of the containment sump water to the RWST. The RWST is located in the plant yard and is vented to the atmosphere. Since this release path represents a bypass of the containment, dose consequences are considered. The concentration of radionuclides in the containment sump water is as modeled above for ECCS leakage. FENOC assumes that containment sump water leaks into the RWST at a rate of 2 gpm starting at about 36 minutes. At 86 minutes, the radioiodines are projected to be released via the RWST vent and continue for 30 days. FENOC assumes that a portion of the iodine dissolved in the backleakge will be retained within the RWST. The time-dependent iodine release fractions used by FENOC are illustrated in Figure 5.3.6-3 of the June 5, 2002, submittal. Values range from about 0.04 at 5178 seconds post-accident, to a minimum value of about 9.0E-8 at 1.2E6 seconds. FENOC assumes that this activity is exhausted without filtration, and that the chemical form of the iodine released is 97 percent elemental and 3 percent organic. The assumptions for the RWST backleakage evaluation reflect the changes in the RWST ECCS switchover setpoint and liquid temperatures associated with the containment conversion.

3.5.6 LOCA Control Room Modeling

Due to the rapid pressure transient, FENOC assumes that a CIB signal is actuated at T=0. Control room isolation is assumed to occur at 77 seconds. Although automatic actuation of CREVS is assumed at 137 seconds, FENOC conservatively assumes in dose calculations that CREVS is actuated by operator manual actions at T=30 minutes. See Section 3.3 above for further discussion of control room modeling.

3.5.7 Summary–LOCA

Details on the assumptions found acceptable to the NRC staff are presented in Table 2 below. The estimated doses for the postulated DBA LOCA were found to be acceptable.

3.6 CREA

This DBA analysis postulates the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected. This failure breeches the reactor pressure vessel head resulting in a LOCA to the containment. The release to the environment is assumed to occur through two separate pathways:

- Release of containment atmosphere (i.e., design leakage)

- Release of RCS inventory via primary-to-secondary leakage through steam generators (SGs)

While the actual doses from a CREA would be a composite of the two pathways, an acceptable dose from each pathway, modeled as if it were the only pathway, would demonstrate that the composite dose would also be acceptable.

FENOC assumed that 10 percent of the fuel rods fail releasing the fission product inventory in the fuel rod gap. It is assumed that 10 percent of the core inventory of iodines and noble gases is in the fuel rod gap. A radial peaking factor of 1.75 was applied. In addition, localized heating is assumed to cause 0.25 percent of the fuel to melt, releasing 100 percent of the noble gases and 25 percent of the iodines contained in the melted fuel to be released to the containment. For the secondary release case, 100 percent of the noble gases and 50 percent of the iodines contained in the melted fuel to be released to the secondary. Because the containment sump pH is not controlled for a CREA, FENOC conservatively assumed that the chemical form of the iodine released to the environment would be 97 percent elemental and 3 percent organic. These assumptions are consistent with RG 1.183 and are acceptable.

For the containment leakage case, the fission products released from the fuel are assumed to be instantaneously and homogeneously mixed in the containment free volume. The containment is projected to leak at its design leakage of 0.1 percent of its contents by weight per day for the first 24 hours and then at 0.05 percent for the remainder of the 30-day accident duration. These leakage assumptions were made in support of a proposed containment conversion for re-analysis of containment pressure transient analyses which has not been approved by the NRC. However, this assumed leakage exceeds that currently provided for by the design basis (i.e., 0.1 percent volume per day for 1 hour). This is a conservative situation. This leakage is not collected and enters the environment without processing.

For the secondary release case, the fission products released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side of the SGs via primary-to-secondary leakage at the technical specification value of 150 gpd per SG for 2500 seconds. The SG tubes remain covered for the duration of the event. An iodine partition factor of 100 was assumed in the SGs. There is no mitigation of the released noble gases. A loss-of-offsite power (LOOP) is conservatively assumed to occur at T=0, rendering the main condenser unavailable. With the main condenser unavailable, the plant is cooled down by releases of steam to the environment via atmospheric dump valves (ADVs) and/or the main steam safety valves (MSSVs). Releases from the SGs will continue for 8 hours, at which time shutdown cooling is initiated and the environmental releases are terminated. Control room response is as described in Section 3.3 above, with the exception that actuation by manual action at 30 minutes, rather than by the CIB signal, is assumed.

Details on the assumptions found acceptable to the NRC staff are presented in Table 2 below. The estimated doses for the postulated CREA were found to be acceptable.

3.7 MSLB

The accident considered is the complete severance of a main steam line outside containment. The radiological consequences of a break outside containment will bound those results from a break inside containment. Thus, only the break outside is considered with regard to dose. The faulted SG will rapidly depressurize and release the initial contents of the SG to the environment. As a result of the rapid depressurization and the resulting high pressure differentials, some SG tubes fail resulting in an increase in the primary-to-secondary leakage (i.e., accident-induced leakage). Releases to the environment continue until the plant is cooled down.

The current licensing basis assumes that CREBAPS is manually actuated at T=30 minutes, followed by the actuation of CREVS 60 minutes later. The proposed configuration calls on manual actuation of CREVS at T=30 minutes. FENOC used a scaling approach to assess the impact of the modified system response on the previously analyzed control room operator doses. No other changes were made to the analyses. Offsite doses were not re-assessed since the proposed changes would not have an impact on these doses. In performing its review, the NRC staff did not rely on the scaling approach but, instead, performed confirmatory analyses using the previously approved licensing basis methodologies, assumptions and regulatory acceptance criteria, adjusted for the modified control room habitability response.

Some significant assumptions used in the current licensing basis analyses which FENOC scaled, and which were the basis for the NRC staff’s confirmatory calculations include: FENOC assumes that the faulted SG boils dry in 30 minutes, releasing the entire liquid inventory and dissolved radioiodines through the faulted steam line to the environment. A LOOP is conservatively assumed to occur when the reactor trips. This LOOP renders the main condenser unavailable. With the main condenser unavailable, the plant is cooled down by releases of steam to the environment via ADVs and/or MSSVs on the unaffected SG until residual heat removal (RHR) system cooling is started; assumed to be 8 hours. Primary-to-secondary leakage is assumed to be 150 gpd to each SG. Based on nondestructive testing results and correlations to destructive testing on representative degraded tubes, FENOC has established the accident-induced leakage at BVPS-1 and 2 to be 14.5 gpm and 2.5 gpm, respectively. No fuel damage is postulated to occur because of an MSLB. FENOC assumes the initial iodine inventory in the RCS and SG to be at the maximum concentrations permitted by the TSs. Two iodine spiking cases are considered. The first assumes that an iodine spike occurred just before the event and the RCS iodine inventory is at the maximum value (for 100 percent power) permitted by the TSs. The second case assumes the event initiates an iodine spike. Iodine is released from the fuel to the RCS at a rate 500 times the normal iodine appearance rate for 8 hours.

Details on the assumptions found acceptable to the NRC staff are presented in Table 2 below. The estimated doses for the postulated MSLB at BVPS-1 and 2 were found to be acceptable.

3.8 LRA at BVPS-1

For this accident, a reactor coolant pump rotor is assumed to seize instantaneously causing a rapid reduction in the flow through the affected RCS loop. A reactor trip will occur, shutting down the reactor. The flow imbalance creates localized temperature and pressure changes in the core. If severe enough, these differences may lead to localized boiling and fuel damage. The radiological consequences are due to leakage of the contaminated reactor coolant to the SGs and from there, to the environment. A LOOP is conservatively assumed to occur when the reactor trips, rendering the main condenser unavailable. With the main condenser unavailable, the plant is cooled down by releases of steam to the environment via ADVs and/or the MSSVs.

The LRA is projected to result in 18.0 percent failed fuel and the release of the associated gap activity. This release is assumed to be instantaneously and homogeneously mixed in the reactor coolant system and transported to the secondary side via SG tube leakage assumed to be at the TS value of 150 gpd for each SG. FENOC assumed that 12 percent of the I-131 inventory of the core was in the fuel rod gap, along with 30 percent of the Kr-85, and 10 percent of all other iodines, and noble gases. FENOC assumed an iodine partitioning factor of 100 in the SGs. The iodine releases from the SGs are assumed to be 97 percent elemental and 3 percent organic. Noble gases are released to the environment without holdup in the SGs. Control room response is as described in Section 3.3 above, with the exception that actuation by manual action at 60 minutes, rather than by the CIB signal is assumed.

Details on the assumptions found acceptable to the NRC staff are presented in Table 2 below. The estimated doses for the postulated LRA at BVPS-1 were found to be acceptable.

3.9 TS Changes

As stated above, BVPS-1 and 2 is served by a single control room that supports both units. Current plant design will automatically isolate the control room and initiate control room pressurization via the CREBAPS upon receipt of a CIB signal from either unit, or a high radiation alarm from the control room area monitors. The above signals will also initiate the time delay relay that delays start of the CREVS for 60 minutes which is intended to correspond to the depletion of the bottled air and the time for the containment to return to sub-atmospheric pressure.

One hour after actuation, the time delay relay will automatically start one of the trains of BVPS-2 CREVS fans. On detection of fan failure, the second train is automatically initiated after a short time delay. In the unlikely event that neither of the BVPS-2 trains can be put in service, operator action may be utilized to initiate the BVPS-1 CREVS. This unlikely scenario is utilized in accident analysis to allow flexibility in taking out a BVPS-2 CREVS train for maintenance.

As a result of this change request, BVPS-1 and 2 design basis will be changed to eliminate the accident mitigation function of CREBAPS and the 60-minute time delay associated with the CREVS. Also, no credit is taken for the capability of the safety-related control room radiation monitors to automatically initiate the CREVS. The BVPS-1 and 2 CIB signals will initiate the BVPS-2 CREVS without a 60-minute time delay. In the event one of the BVPS-2 trains is out of service, and the second train fails to start, manual operator action will be utilized to initiate the BVPS-1 CREVS.

3.9.1 The licensee has proposed the following changes to the TS LCOs, surveillance requirements (SRs) and Action Statements.

3.9.1.1 TSs 3.7.7.1 (BVPS-1) and 3.7.7 (BVPS-2) for the Control Room Emergency Habitability Systems are revised to remove the operability requirements for the CREBAPS from both the BVPS-1 and BVPS-2 TSs. Specifically, LCO item b is deleted (and the associated footnote) along with the Action Statements and surveillance requirements related to operability of the CREBAPS.

Technical Specification 3.7.7.1 BVPS-1, item b. states:

> b. Five bottled air pressurization subsystems consisting of two bottles per subsystem are OPERABLE**, and

This LCO is being deleted.

TS 3.7.7.1, BVPS-1, associated footnote ** states:

> ** The air bottles may be isolated for up to 8 hours for performance of instrumentation and control systems testing.

This footnote is being deleted.

TS 3.7.7, BVPS-2, item b. states:

> b. A bottled air pressurization system comprised of 5 subsystems with two bottles in each subsystem.*

This LCO is being deleted.

TS 3.7.7, BVPS-2, associated footnote * states:

* The air bottles may be isolated for up to 8 hours for performance of instrumentation and control systems testing.

This footnote is being deleted.

3.9.1.2 ACTION STATEMENTS Relating to the Above LCO's

TS 3.7.7.1, BVPS-1, Items b. and b.1 ACTION STATEMENTS state:

b. With one bottled air pressurization subsystem inoperable, restore five bottled air pressurization subsystems to OPERABLE within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b.1 With less than four bottled air pressurization subsystems OPERABLE, the requirements of 3.0.3 are applicable and movement of irradiated fuel assemblies and movement of fuel assemblies over irradiated fuel assemblies shall be suspended.

Both ACTION STATEMENTS are being deleted.

TS 3.7.7, BVPS-2, MODES 1, 2, 3, and 4 ACTION STATEMENT states:

With one train of the pressurization filtration unit, or one subsystem of the bottled air pressurization system, or one of two isolation dampers in series inoperable, restore the system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The phrase "*or one subsystem of the bottled air pressurization system*" is being deleted.

ACTION STATEMENT a., following the statement:

During movement of recently irradiated fuel assemblies, and during movement of fuel assemblies over recently irradiated fuel assemblies:

states:

a. With one train of the pressurization filtration unit, or one subsystem of the bottled air pressurization system, or one of two isolation

> > dampers in series inoperable, restore the inoperable system to OPERABLE status within 7 days or suspend all operations involving movement of recently irradiated fuel assemblies and movement of fuel assemblies over recently irradiated fuel assemblies.

The phrase "*or one subsystem of the bottled air pressurization system*" is being deleted.

ACTION STATEMENT b., following the statement:

> During movement of recently irradiated fuel assemblies, and during movement of fuel assemblies over recently irradiated fuel assemblies:

states:

> b. With both trains of the pressurization filtration unit, or more than one subsystem of the bottled air pressurization system, or two of two isolation dampers in series inoperable, suspend all operations involving movement of recently irradiated fuel assemblies and movement of fuel assemblies over recently irradiated fuel assemblies.

The phrase "*or more than one subsystem of the bottled air pressurization system*" is being deleted

The licensee stated that these items are to be deleted from the TSs or modified as shown above and that the subsequent paragraphs in the TSs are to be renumbered appropriately.

The NRC staff review noted that the CREBAPS which is part of the Control Room Emergency Air Clean Up and Pressurization System is currently required to have a TS LCO to satisfy 10 CFR 50.36, Criterion 3, "[a] structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The licensee states that it no longer credits the CREBAPS in its analysis for any of the DBAs. The NRC staff confirmed that the CREBAPS is not credited in the review of the DBA analyses. Since the bottled air pressurization system is no longer considered to be on the primary success path because its actuation is not credited in the DBA, the NRC staff concludes that Criterion 3 no longer applies.

The CREBAPS does not impose a process variable, design feature, or operating restriction on any DBA analysis that is an initial condition. Thus, Criterion 2 of 10 CFR 50.36 also does not apply.

There is insufficient operating experience to conclude this system is significant to public health and safety. Also, there has not been a probabilistic risk assessment that indicated this system was significant. As such, Criterion 4 of 10 CFR 50.36 does not apply.

The NRC staff also reviewed whether the requirements of Criterion 1 apply and determined that this system does not relate to installed instrumentation that is used to detect abnormal degradation of the RCS. Thus, Criterion 1 of 10 CFR 50.36 does not apply.

The NRC staff concludes that the proposed TS changes to delete the TS requirements for the CREBAPS conforms to the criteria in 10 CFR 50.36. The NRC staff has determined that the requirements can be removed from the BVPS-1 and 2 TSs with reasonable assurance that the public health and safety is adequately protected.

3.9.2 The licensee has also proposed the following changes to the TS SRs.

3.9.2.1 SRs 4.7.7.1.1.c.1 and 4.7.7.1.2.c.1 for control room emergency habitability systems are revised by changing the BVPS-1 and 2 emergency ventilation filtration systems' allowable penetration acceptance criteria. The by-pass leakage testing requirements are changed from <1 percent to <0.05 percent for the HEPA and charcoal filters in the BVPS-1 TSs.

[Note: 'penetration' is the amount of the test gas (methyl iodide) that passes through the filter under test conditions. BVPS-1 TSs are being changed to be consistent with BVPS-2 TSs. BVPS-2 TSs are not being changed by this paragraph.]

BVPS-1, SR 4.7.7.1.1 c. 1 states:

1. Verifying that the filtration system satisfies the in place penetration and by-pass leakage testing acceptance criteria of less than 1% when tested in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 800 - 1000 cfm.

This SR is being changed to the following:

1. Verifying that the filtration system satisfies the in place penetration and by-pass leakage testing acceptance criteria of less than 0.05% when tested in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 800 - 1000 cfm.

BVPS-1, SR 4.7.7.1.2 c.1 states:

1. Verifying that the filtration system satisfies the in place penetration and by-pass leakage testing acceptance criteria of less than 1%

when tested in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 800 - 1000 cfm.

This SR is being changed to the following:

1. Verifying that the filtration system satisfies the in place penetration and by-pass leakage testing acceptance criteria of less than 0.05% when tested in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 800 - 1000 cfm.

The NRC staff review noted that the reduction in the in place penetration and by-pass leakage testing acceptance criteria from less than 1 percent to less than 0.05 percent is a conservative change in the test requirements. The licensee credited the filter with 99 percent efficiency for aerosols. The less than 0.05 percent criteria results in a safety factor of 2. This is also consistent with the NRC staff position as outlined in RG 1.52, Revision 3. This change enhances system performance and makes the requirements for BVPS-1 identical to BVPS-2. Thus, this change is acceptable.

3.9.2.2 BVPS-1, SR 4.7.7.1.2 d.2 states:

2. Verifying that on a Containment Isolation Phase B/Control Room High Radiation test signal from either Unit, the system automatically closes all the series isolation ventilation system dampers which isolate the combined control room from the outside atmosphere and the system automatically starts 60 minutes later and supplies air to the control room through the HEPA filters and charcoal adsorber banks.

This SR is being changed to the following:

2. Verifying that on a Containment Isolation Phase B/Control Room High Radiation test signal from either Unit, the system automatically closes all the series isolation ventilation system dampers which isolate the combined control room from the outside atmosphere and the system automatically starts and supplies air to the control room through the HEPA filters and charcoal adsorber banks.

The NRC staff review noted that the 60-minute delay time in the start of the CREVS was based on the operation of the CREBAPS being operational during the first hour of a DBA. Since the CREBAPS will not be credited with operation, the CREVS must start automatically at the onset of the appropriate signals. This change is necessary as part of the elimination of the

CREBAPS and assures that the control room operators receive a clean source of supply air to pressurize the control room and minimize inleakage. This change is acceptable because it enhances the habitability of the control room and thus provides a measure of protection for the operator with no adverse impact on public health or safety. This change is also consistent with the requirements of GDC 19.

3.9.2.3 BVPS-1, SR 4.7.7.2 states:

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 10 bottles of air each pressurized to at least 1825 psig and by verifying that the system solenoid operated valves are powered from an operable emergency bus.

b. At least once per 18 months by verifying that:

1. A control room high radiation/containment phase B isolation test signal from either Unit will initiate system operation.

2. Upon a partial discharge test using four out of five bottled air subsystems the system will supply $\leq$ 1000 cfm of air and pressurize the control room to $\geq$ 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

This SR is being deleted.

The NRC staff determined that the CREBAPS is no longer required for accident mitigation and is therefore not on the primary success path for any DBA. As such, the NRC staff concurs that the need for maintaining these TS SRs has been eliminated. The proposed changes in the SRs stated above are acceptable and are consistent with the maintenance of public health and safety.

3.9.2.4 BVPS-2, SR 4.7.7.1b states:

b. At least once per 31 days by:

1. Initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for 15 minutes with the heaters in operation.

2. Verifying that the bottled air pressurization system contains a minimum of 10 bottles of air each pressurized to at least 1825 psig and that each solenoid operated valve is powered from an operable emergency bus.

This SR is being changed to the following:

b. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for 15 minutes with the heaters in operation.

The NRC staff determined that the CREBAPS is no longer required for accident mitigation and is therefore not on the primary success path for any DBA. As such, the NRC staff concurs that the need for maintaining this TS SR has been eliminated. The proposed deletion of the SRs in item b. 2. above is acceptable and is consistent with the maintenance of public health and safety. The changes in item b. 1. (renumbered item b.) are editorial and acceptable.

3.9.2.5 BVPS-2, SR 4.7.7.1 e.2 states:

2. Verifying that on a Containment Isolation Phase B/Control Room High Radiation test signal, the system automatically closes all the series isolation ventilation system dampers which isolate the control room from the outside atmosphere and the system automatically starts 60 minutes later and supplies air to the control room through the HEPA filters and charcoal adsorber banks.

This SR is being changed to the following:

2. Verifying that on a Containment Isolation Phase B/Control Room High Radiation test signal, the system automatically closes all the series isolation ventilation system dampers which isolate the control room from the outside atmosphere and the system automatically starts and supplies air to the control room through the HEPA filters and charcoal adsorber banks.

The NRC staff review noted that the 60-minute delay time for the start of the CREVS was based on the operation of the CREBAPS during the first hour following the initiation of a DBA. Since the CREBAPS will not be credited with operation during any DBA, the CREVS must start automatically at the onset of the appropriate signals. This change is necessary as part of the elimination of the CREBAPS and assures that the control room operators receive a clean source of supply air to pressurize the control room and minimize inleakage. This change is acceptable because it enhances the habitability of the control room and thus provides a measure of protection for the operator without adverse impact on public health or safety. This change is also consistent with the requirements of GDC 19.

3.9.2.6 BVPS-2, SRs 4.7.7.1 e.6 and e.7 state:

6. Verifying that a control room high radiation/containment phase B isolation signal will initiate operation of the bottled air pressurization system.

7. Verifying by a partial discharge test from four out of five sub-systems of the bottled air pressurization system at a discharge flow of less than 1000 cfm that the bottled air pressurization system will pressurize the control room to ≥ 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

These two SRs are being deleted.

The NRC staff determined that the CREBAPS is not required for accident mitigation and is therefore not on the primary success path for any DBA. As such, the NRC staff concurs that the need for maintaining these TS SRs has been eliminated. The proposed changes in the SRs stated above are acceptable and are consistent with the maintenance of public health and safety.

3.9.2.7 TS 3.3.3.1 (BVPS-1 only) is revised by removing reference to the CREBAPS in SR Table 4.3-3, footnote ###, for the control room area radiation monitors.

Table 4.3-3, “Radiation Monitoring Instrumentation Surveillance Requirements,” footnote ### states:

Control Room intake and exhaust isolation dampers and CREBAPS solenoid valves are not actuated.

This footnote is being changed to the following:

Control Room intake and exhaust isolation dampers are not actuated.

The NRC staff determined that the CREBAPS is no longer necessary for accident mitigation and is therefore not on the primary success path for any DBA. As such, the NRC staff concurs that the need to reference CREBAPS in the footnote has been eliminated. The proposed changes in the SRs stated above are acceptable and are consistent with the maintenance of public health and safety.

3.10 Design-Basis Change

The licensee is requesting a design-basis change which would allow the facility to make the CREBAPS system nonoperational and permit the removal of the system. In the above Sections of this safety evaluation, it was determined that there were no requirements for TS control over this system based on the review of 10 CFR 50.36. The primary consideration was the adequacy of the analyses that evaluated the radiological consequences of DBAs. The NRC staff found the results to be in compliance with the NRC guidelines.

In this section, the impact of removal of the system from the design basis of the facility is considered. Since control room pressurization is credited in the DBA analysis, the NRC staff reviewed the CREVS to assure adequate pressurization capability exists with the CREBAPS removed. Since the CREBAPS is identified in the UFSAR as being in operation for toxic gas and smoke from a plant fire mitigation, the NRC staff reviewed the toxic gas and smoke mitigation capabilities with the CREBAPS removed.

3.10.1 Adequate and Redundant Control Room Pressurization Systems

Currently, during the first 60 minutes, the CREBAPS provides control room pressurization. There are two trains of CREBAPS, thus redundancy is provided. Subsequent to the proposed design-basis change, the CREBAPS will not be functional and may be physically removed. The 60-minute delay on the start of the two BVPS-2 CREVS systems will be removed such that one CREVS will start immediately and the other CREVS will start if the first unit fails to start. The single BVPS-1 CREVS is a manual system that requires a manual opening of intake and exhaust dampers and a manual start of the fan. The 3 CREVS trains serve the common control room envelope. Any of the 3 CREVS trains is capable of achieving the required pressurization.

The BVPS-1 TSs require 2 out of 3 CREVS trains be operational. The worst case would be a situation where one automatically started BVPS-2 CREVS train and one manually started BVPS-1 CREVS train were operating. The remaining BVPS-2 CREVS train is assumed to be nonoperational. In the event that a BVPS-2 CREVS train failed to start on a CIB signal, the operator would have to manually start the BVPS-1 CREVS train. This would require an operator to be dispatched with the appropriate respiratory equipment to the ventilation room to open intake and exhaust dampers, start the fan, and verify operation. The licensee assumed that this action could be completed in 30 minutes and has stated that it verified that these actions could be taken in 20 minutes during an unannounced drill. The licensee assumed 30 minutes to achieve pressurization in its DBA analyses and satisfied the requirements of GDC 19 with respect to acceptable control room dose. As such, the licensee has demonstrated that the BVPS-1 TS requirements are sufficient to support the assumptions in the DBA analysis, that control room pressurization is achieved in an acceptable time frame with either the automatically initiated CREVS trains or a combination of one automatically initiated CREVS train and one manually initiated CREVS train.

The BVPS-2 TSs require 2 out of 2 CREVS trains to be operational. The NRC staff has interpreted this to mean the 2 automatically initiated CREVS trains are assigned to BVPS-2 and that the manual CREVS of BVPS-1 is not considered in satisfying this requirement. This is more restrictive than the BVPS-1 TS requirement and is acceptable for redundancy and pressurization purposes.

The NRC staff noted some inconsistencies between the BVPS-1 and 2 TSs. These inconsistencies are unaffected by the removal of the CREBAPS system. The licensee has noted the differences and stated in its June 24, 2003, letter that “the ongoing Beaver Valley Improved Standard Technical Specification Conversion Project will address the differences between the units’ Technical Specifications by combining the specifications for each unit into a single set of specifications addressing both units.”

3.10.2 Toxic Gas and Smoke

The BVPS-2 UFSAR, Table 9.4-4, states that the CREBAPS is activated for toxic gas control and for plant fire smoke control. Actuation of the CREBAPS system would provide pressurization of the control room during the first hour from a clean, uncontaminated source and minimize unfiltered inleakage into the control room.

In response to an NRC request for additional information (RAI), the licensee in its June 24, 2003, letter stated that Table 9.4-4 in the BVPS-2 UFSAR was incorrect “because the control room emergency bottled air pressurization system (CREBAPS) is not used for these two scenarios” and that the BVPS-1 and 2 corrective action program “will track the appropriate corrections to the UFSAR and ensure they are completed in a timely manner.”

3.10.2.1 Toxic Gas

The licensee stated in its June 24, 2003, letter that the previous onsite storage of gaseous chlorine used for water treatment has been discontinued and thus there is no onsite chlorine toxic gas threat to be considered. The licensee also stated that a “complete evaluation of all other toxic gas hazards from onsite, offsite, and transportation sources was performed and accepted by the NRC prior to the onsite chlorine being removed.” This is documented in BVPS-2 UFSAR, Section 2.2.3.1.2. This evaluation determined “that the probability of a toxic chemical spill resulting in unacceptable exposures was less than the NRC design basis criteria and thus did not have to be included in the plant design basis.” The licensee also stated that current procedures require the isolation of the control room in the event of toxic gas. The CREBAPS is not used in the current procedure.

3.10.2.2 Smoke From a Plant Fire

The licensee stated in its June 24, 2003, letter that for a plant fire outside of the control room, that the control room HVAC (heating, ventilation and air conditioning) system would be placed in a 100 percent recirculation mode with outside air intakes and exhausts isolated with the CREVS fans off. The CREVS fans which would pressurize the control room “are not used during a fire, because the smoke can quickly overcome the HEPA filter and render the system inoperable.” The licensee stated that the “300 cfm of unfiltered inleakage is associated with operation of the normal control room ventilation air conditioning units” [during periods of control room isolation] and also states that “reliance is placed on the use of self-contained breathing apparatus masks, if necessary. However, it is unlikely that masks will be necessary, since 300 cfm inleakage is small compared to the volume of the control room envelope, 173,000 cubic feet.”

The NRC staff review of the design-basis change request and concludes that there would be sufficient capability to provide control room pressurization with the existing CREVS to replace the function currently served by the CREBAPS system. The NRC staff also accepts the licensee's assessment of mitigation of toxic gas and smoke from a plant fire which does not rely upon the CREBAPS system. Therefore, the proposed design-basis change is acceptable.

3.11 Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by FENOC to assess the radiological impacts of the proposed changes related to removing requirements for the CREBAPS at BVPS-1 and 2. The NRC staff finds that FENOC used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by FENOC to the applicable criteria identified in Section 2.0. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. Therefore, the proposed changes are acceptable with regard to the radiological consequences of postulated design-basis accidents.

This licensing action is considered a selective implementation of the AST. With this approval, the selected characteristics of the AST and TEDE criteria become the design basis for the DBA LOCA and CREA events. This approval is limited to this specific implementation. Subsequent modifications based on the selected characteristics incorporated into the design basis by this action may be possible under the provisions of 10 CFR 50.59. However, changes to previously approved AST characteristics require prior NRC staff approval under 10 CFR 50.67. The selected characteristics of the AST and the TEDE criteria may not be extended to other aspects of the plant design or operation without prior NRC review under 10 CFR 50.67. All future DBA LOCA and CREA radiological analyses performed to demonstrate compliance with regulatory requirements shall address the selected characteristics of the AST and the TEDE criteria as described in the BVPS-1 and 2 design basis.

Since these analyses were performed at a power level of 2918 MWt (100.6 percent of 2900 MWt), the NRC staff finds that the radiological consequences of the DBA LOCA and CREA events would remain bounding up to a licensed thermal power of 2900 MWt. However, the approval of this amendment does not constitute authority to operate above the current licensed rated thermal power.

TABLE 2

ANALYSIS ASSUMPTIONS

Assumptions Common to One or More Analyses

Dose conversion factors - ICRP30

Offsite breathing rate, m^3/sec

0-8 hours	3.47E-4
8-24 hours	1.75E-4
24-720 hours	2.32E-4

Control room volume, ft^3 173,000

Normal ventilation make-up flow, cfm 500

Control room HVAC system

Filtered air make-up, cfm	600-1030
Recirculation, cfm	0
Isolation mode unfiltered inleakage, cfm	300
Pressurization mode unfiltered inleakage, cfm	30
Intake filter efficiency,%	
Aerosols	99
Elemental/organic	98

Control room breathing rate, m^3/sec 3.47E-4

Control room occupancy factors

0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4

Offsite χ/Q, sec/m^3		BVPS-1	BVPS-2
EAB:	0-2 hr	1.04E-3	1.25E-3
LPZ:	0-8 hr	6.04E-5	6.04E-5
	8-24 hr	4.33E-5	4.33E-5
	24-96 hr	2.10E-5	2.10E-5
	96-720 hr	7.44E-6	7.44E-6

ERF room volume, ft^3	478,610
Normal ventilation make-up flow, cfm	3800 +/- 10%
Normal ventilation filter efficiency (HEPA),%	99
Normal operation unfiltered inleakage, cfm	2090

ERF HVAC system

Filtered air make-up, cfm	0
Recirculation, cfm	3800 +/-10%
Isolation mode unfiltered inleakage, cfm	910
Isolation delay, min	30
Recirculation filter efficiency,%	Not credited

ERF breathing rate, m^3/sec 3.47E-4

ERF occupancy factors

0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4

Assumptions for LOCA Analyses BVPS-1 and 2

Reactor power (includes 0.6% uncertainty), Mwt 2918

Onset of gap release phase, sec 30

Core release fractions and timing–Containment atmosphere

Duration, hrs	0.5000E+00	0.1300E+01
Noble Gases:	0.5000E-01	0.9500E+00
Iodine:	0.5000E-01	0.3500E+00
Cesium:	0.5000E-01	0.2500E+00
Tellurium:	0.0000E+00	0.5000E-01
Sr, Ba:	0.0000E+00	0.2000E-01
Noble Metals:	0.0000E+00	0.2500E-02
Cerium:	0.0000E+00	0.5000E-03
Lanthanum:	0.0000E+00	0.2000E-03

Iodine species distribution

Elemental	0.95
Organic	0.0485
Particulate	0.0015

Containment Leakage Pathway

Containment free volume, ft^3	1.75E6
Sprayed, ft^3	1.102E6
Unsprayed, ft^3	6.48E5

Containment release

0-24 hours, %/day	0.1
24-720 hours, %/day	0.05

Containment spray removal

Containment sprayed fraction,%	63
Credited spray start, sec	722
Spray termination, days	4
Sprayed/unsprayed mixing rate, unsprayed volume/hour (cfm)	2 (21580)
Maximum DF	200

Aerosol removal rates, 1/hr *(values extracted from licensee graph)*

Period		Sprayed	Unsprayed
0	-1800s	4	0.004
1800	-2400	30	0.030
2400	-3000	40	0.056

3000 -3600	46	0.062
3600 -4800	53	0.067
4800 -5400	56	0.072
5400 -6600	45	0.075
6600 -7200	10	0.080
7200 -9000	4	0.090
9000 -12600	2	0.093
12600-18000	1	0.092

Elemental iodine removal rate, 1/hr	Sprayed aerosol $\leq \lambda \leq$ 20
Elemental iodine removal by plateout, 1/hr	2
Release point	Containment outer wall SLCRS vent
SLCRS filtration	Not credited
Control room isolation, sec	77
Control room pressurization, sec	1800

ECCS Leakage Pathway

Start of ECCS leakage, minutes	5
ECCS leak rate (includes 2x multiplier), ml/hr	11400
Duration of release, days	30
Containment sump volume, ft^3 (lbm)	
5-30 min	9800 (5.77E5)
0.5-2 hr	28,600 (1.71E6)
2 hr-30 day	65,600 (4.01E6)
Fraction of core inventory iodine in sump	0.4
Iodine flash fraction	0.1
Chemical form release fractions	
Elemental	0.97
Organic	0.03
Release pathway	via SLCRS Vent (unfiltered)

Release from RWST

Containment sump water backleakage to RWST (includes 2x multiplier), gpm (cfm)	2 (0.267)
Start of backleakage, seconds	2186
Start of RWST venting, seconds	5178
Iodine release rate, % per day	

Seconds	Rate
2186 - 7500	4.0
7500 - 10000	1.4
10000 - 20000	0.9
20000 - 30000	0.5
30000 - 150000	0.12
150000 - 300000	5.0E-2
300000 - 600000	5.0E-3
600000 - 1500000	4.0E-4
1500000 - 2500000	1.5E-4

Release pathway	RWST Vent

Release from Containment Vacuum System

Source term	RCS T/S Activity
Chemical form release fractions	
Elemental	0.97
Organic	0.03
Release rate, scfm	2200
Release duration, sec	5
Release point	Containment wall SLCRS vent

Assumptions for MSLB Analyses

	BVPS-1	BVPS- 2
Initial RCS activity, µCi/gm dose equivalent I-131	0.1	0.35
Initial secondary activity, µCi/gm dose equivalent I-131	0.05	0.1
Pre-incident iodine spike activity, µCi/gm dose equivalent I-131	6.0	21.0
Co-incident spike appearance rate, Ci/hr		
I-131	1166	4553
I-132	1146	4629
I-133	2008	7897
I-134	1275	5205
I-135	1517	6037
RCS letdown flow rate, gpm	120.0	135.0
RCS letdown demineralizer efficiency	1.0	1.0
RCS mass, lbm	329,500	388,700
Iodine spike duration, hrs	4	4
Primary-to-secondary leakage		

Pre-event @ SG, gpd	150	150
Faulted SG, gpm	14.5	2.5
To @unaffected SGs, gpd	150	150
Primary to secondary leakage duration, hours	8	8
SG mass, lbm		
Liquid	162,900	162,900
Vapor	6400	6400
Steam release from faulted SG, lbm		
0-30 minutes, lbm	170,050	169,306
30 minutes-8 hr,	21,200	9779
Steam release from unaffected SGs, lbm		
0-2 hours	336,776	336,776
2-8 hours	705,393	705,393
Steam partition coefficient in SGs		
Faulted SG	1.0	1.0
Unaffected SG <1 hour	1.0	1.0
Unaffected SG >1 hour	0.01	0.01
Control room isolation delay, min	30	30

Assumptions for CREA Analyses for BVPS-1 and 2

Reactor power (includes 0.6percent uncertainty), MWt		2918
Core inventory		June 5, 2002, submittal, Table 5.3.3-1
Containment free volume, ft^3		1.75E6
Fraction of rods that exceed DNB		0.10
Gap fraction, all nuclide groups		0.10
Fraction of rods that exceed DNB that experience melt		0.0025
Melt isotopic composition	Containment	SG
Noble gases	1.0	1.0
Iodine	0.25	0.5
Radial peaking factor	1.75	
Iodine species fraction	Containment	SG
Particulate/aerosol	95	0
Elemental	4.85	97
Organic	0.15	3
Containment release, %/day		
0-24 hours		0.1
24-720 hours		0.05
Duration of release, days		30

RCS mass, lbm	340,711
Primary-to-secondary leakrate, gpd @SG	150
Terminate primary-to-secondary leakrate, sec	2500
SG mass (@), lbm	99,217
Steaming releases (total), lbs/sec	
0-150 sec	900
150-300 sec	300
300-2500 sec	150
2500 sec-8 hours	29.5
Release duration, hours	8
Steam partition coefficient	0.01
Containment surface/SLCRS vent χ/Q used for containment leakage	
MSSV/ADV χ/Q used for SG releases	
Control room isolation, minutes	30

Assumptions for BVPS-1 LRA Analyses

Core inventory	UFSAR Table 14B.1
Fraction of fuel with defects	0.18
Fraction of core inventory in gap	
Kr-85	0.3
I-131	0.12
Others	0.1
Initial RCS activity, µCi/gm dose equivalent I-131	0.1
Initial secondary activity, µCi/gm dose equivalent I-131	0.05
Co-incident spike appearance rate, Ci/hr	
I-131	1166
I-132	1146
I-133	2008
I-134	1275
I-135	1517
RCS mass, lbm	345800
Iodine spike duration, hrs @SG	4
Primary-to-secondary leakage duration, hours	8
SG mass, lbm	
Liquid @	93481
Vapor	6388
Steam release from SGs, lbm	

0-2 hours	443,878
2-8 hours	793,644
Steam partition coefficient in SGs	0.01
Control room isolation delay, min	60

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (67 FR 75876). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: S. LaVie
E. Forrest
A. Drozd
T. Colburn

Date: September 10, 2003

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NOS. 50-334 AND 50-412

NOTICE OF PARTIAL WITHDRAWAL OF APPLICATION FOR

AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of FirstEnergy Nuclear Operating Company, et al. (the licensee), to partially withdraw its application for proposed amendment to Facility Operating License Nos. DPR-66 and NPF-73 for the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2), located in Beaver County, Pennsylvania.

The request to review related Updated Final Safety Analysis Report (UFSAR) pages submitted with the licensee's February 14, 2003, submittal was withdrawn in its March 31, 2003, submittal. The licensee's request to modify TS 3.6.2.2, "Containment Recirculation Spray System," by reducing the required recirculation spray heat exchanger minimum river/service water flow rate in surveillance requirement (SR) 4.6.2.2.e.3, was withdrawn by its March 19, 2003, letter. The changes were no longer necessary as Amendment Nos. 252 and 132 for BVPS-1 and BVPS-2, respectively, relocated SR 4.6.2.2.e.3 to each unit's UFSAR.

The licensee's requested changes related to conversion of the BVPS-1 and 2 containments from subatmospheric to atmospheric operating conditions was withdrawn by its September 5, 2003, submittal.

The Commission had previously issued a Notice of Consideration of Issuance of Amendment published in the FEDERAL REGISTER on December 10, 2002 (67 FR 75876). However, by letters dated March 19 and 31, and September 5, 2003, the licensee withdrew the portions of the proposed changes as stated above.

For further details with respect to this action, see the application for amendment dated June 5, 2002, as supplemented August 19 and December 2, 2002, and January 30, February 14, March 19 and 31, June 6 and 24, and September 5, 2003. The licensee's letters dated March 19 and 31, and September 5, 2003, withdrew a portion of the application for license amendment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams/html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, or 301-415-4737 or by email to pdr@nrc.gov.

Dated at Rockville, Maryland, this 10th day of September 2003.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation