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June 2, 2004
L-04-073

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
License Amendment Request Nos. 317 and 190

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the above licenses in the form of changes to the Beaver Valley Power Station (BVPS) Technical Specifications. The proposed changes will revise the Technical Specifications to permit each unit to be operated with an atmospheric containment design. Following issuance of the requested amendments, the containment for each of the Beaver Valley Power Station (BVPS) units will be operated at slightly sub-atmospheric conditions, in accordance with the proposed Technical Specification changes, and the containment design basis will be changed from being sub-atmospheric within 60 minutes following a design basis accident (DBA) to containment pressure being less than 50 % of peak calculated containment pressure (Pa) within 24 hours following a DBA.

This submittal also requests NRC approval of changes to the methodologies used in the containment analyses. The methodology changes are described in detail in the attached license amendment request and consist of the use of the Modular Accident Analysis Program - Design Basis Analysis (MAAP-DBA) computer code for the BVPS containment integrity analysis and changes to mass and energy calculation methodologies. A pre-application report, describing the MAAP-DBA computer code, was provided to the NRC by Reference 1.

This submittal has two enclosures and five attachments. Enclosure 1 is the FENOC evaluation of the proposed changes. The proposed Technical Specification changes are provided in Attachments A-1 and A-2 for Unit Nos. 1 and 2, respectively. The changes proposed to the Technical Specification Bases are provided in Attachments B-1 and B-2 for Unit Nos. 1 and 2, respectively. The changes proposed to the Licensing Requirements Manual (LRM) are provided in Attachments C-1 and C-2 for Unit Nos. 1 and 2, respectively. The Technical Specification Bases and LRM changes are provided for information only. Attachment D provides a list of commitments associated with this request. Enclosure 2 contains a report describing the revised containment and radiological analyses conducted to support the proposed changes.

The BVPS review committees have reviewed the proposed changes. The changes were determined to be safe and do not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analysis and no significant hazard evaluation.

FENOC requests approval of the proposed amendments by June, 2005. However, since a number of the proposed Technical Specification changes require a plant outage to implement, FENOC requests the following implementation periods. The Unit 1 amendment shall be implemented prior to the first entry into Mode 4 during plant startup from the 1R17 refueling outage. The Unit 2 amendment shall be implemented prior to the first entry into Mode 4 during plant startup from the 2R12 refueling outage. Refueling outage 1R17 is planned for the spring of 2006 and refueling outage 2R12 is planned for the fall of 2006.

If there are any questions concerning this matter, please contact Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement at 724-682-5284.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 2, 2004.

Sincerely,

L. William Pearce

References:
1. FENOC letter L-03-188, "Containment Conversion Pre-application Report," November 24, 2003.

Enclosures:
1. FENOC Evaluation of the Proposed Changes
2. Beaver Valley Power Station Containment Conversion Licensing Report - May 2004

Attachments:

A-1	Unit 1 Proposed Technical Specification Changes
A-2	Unit 2 Proposed Technical Specification Changes
B-1	Unit 1 Proposed Technical Specification Bases Changes
B-2	Unit 2 Proposed Technical Specification Bases Changes
C-1	Unit 1 Proposed Licensing Requirements Manual Changes
C-2	Unit 2 Proposed Licensing Requirements Manual Changes
D	List of Regulatory Commitments

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

ENCLOSURE 1
FENOC Evaluation of the Proposed Changes

Beaver Valley Power Station
License Amendment Requests
317 (Unit 1) and 190 (Unit 2)

Subject: Application to Permit Beaver Valley Power Station Unit Nos. 1 and 2 Operation with an Atmospheric Containment Design.

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Attachments

1.0 DESCRIPTION

This is a request to amend Operating Licenses DPR-66 (Beaver Valley Power Station Unit 1) and NPF-73 (Beaver Valley Power Station Unit 2).

The proposed changes will revise the Operating Licenses to permit each unit to be operated with an atmospheric containment design. The proposed changes reflect revised containment integrity and radiological analyses, as documented in Enclosure 2, "Beaver Valley Power Station Containment Conversion Licensing Report", dated May 2004.

Following issuance of the requested amendments, the containment for each of the Beaver Valley Power Station (BVPS) units will be operated at slightly sub-atmospheric conditions, in accordance with the proposed Technical Specification changes, and the containment design basis will be changed from being sub-atmospheric within 60 minutes following a design basis accident (DBA) to containment pressure being less than 50 % of peak calculated containment pressure (Pa) within 24 hours following a DBA.

2.0 PROPOSED CHANGES

The proposed Technical Specification changes, which are submitted for NRC review and approval, are provided in Attachments A-1 and A-2 for Units 1 and 2 respectively. The changes proposed to the Technical Specification Bases are provided in Attachments B-1 and B-2 for Units 1 and 2 respectively. The changes proposed to the Licensing Requirements Manual (LRM) are provided in Attachments C-1 and C-2 for Units 1 and 2 respectively. The proposed Technical Specification Bases and LRM changes do not require NRC approval. The Beaver Valley Power Station Technical Specification Bases Control Program controls the review, approval and implementation of Technical Specification Bases changes. The BVPS Licensing Document Control Program controls the review, approval and implementation of LRM changes. The Technical Specification Bases and LRM changes are provided for information only. Attachment D provides a list of commitments associated with this License Amendment Request (LAR).

The proposed changes to the Technical Specifications, Technical Specification Bases and LRM have been prepared electronically. Deletions are shown with a strike-through and insertions are shown double-underlined. This presentation allows the reviewer to readily identify the information that has been deleted and added.

To meet format requirements the Index, Technical Specifications, Bases, and LRM pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

2.1 Pending LARs

The changes being proposed in this submittal assume the approval of a recently submitted BVPS LAR, approval of which is expected prior to the approval of this request. Therefore, this request includes the Technical Specification changes from the pending LAR that are germane to this submittal. The applicable pending LAR number appears on the page being changed by this request. The applicable pending LAR is 315 (Unit 1)/188 (Unit 2).

The attached markups for Technical Specification 3.6.3.1 incorporate changes proposed in pending LAR 315 (Unit 1)/188 (Unit 2). The LAR was submitted by FirstEnergy Nuclear Operating Company (FENOC) letter L-03-146 dated October 17, 2003 (Reference 1). The LAR revises Technical Specification 3.6.3.1 to incorporate changes contained in TSTF-30, "Extend the Completion Time for inoperable isolation valve to a closed system to 72 hours" and to provide greater consistency with NUREG-1431, "Standard Technical Specifications - Westinghouse Plants", Revision 2, April 2001. The changes that are proposed in LAR 315/188 do not impact containment conversion. The changes proposed to Technical Specification 3.6.3.1 proposed in this submittal are those that are necessary for containment conversion.

2.2 Proposed Changes

Changes to the following Technical Specifications (TS) are being proposed to allow each unit to be operated as an atmospheric containment design.

Affected Technical Specifications			
No.	**Unit 1**	**Unit 2**	**Title**
1	3.3.2.1	3.3.2.1	Engineered Safety Features Actuation System Instrumentation
1a	Table 3.3-3	Table 3.3-3	Items 1.c [Containment Pressure-High], 2.c [Containment Pressure-High-High], 3.b.3 [Containment Pressure-High-High] and 4.c [Containment Pressure-Intermediate-High-High]
1b	Table 3.3-3	Table 3.3-3	Item 1.1.c, [Refueling Water Storage Tank Level-Low (Unit 1)] Item 1.1.b, [Refueling Water Storage tank Level-Extreme Low (Unit 2)].
1c	Tables 3.3-3 & 4.3-2	N/A	Item 1.1.d [Refueling Water Storage Tank Level – Auto QS Flow Reduction].
2	3.6.1.3	3.6.1.3	Containment Air Locks
	6.17	6.17	Containment Leakage Rate Testing Program
3	3.6.1.4	3.6.1.4	Internal Pressure
4	3.6.1.5	3.6.1.5	Air Temperature
5	3.6.2.3	N/A	Chemical Addition System
6	3.6.3.1	3.6.3.1	Containment Isolation Valves

The following provides a description of the proposed changes and a basis for the change.

Change Number 1a

In this change the allowable values for containment pressure high, containment pressure high-high, and containment pressure intermediate high-high (items 1.c, 2.c, 3.b.3 and 4.c) in Table 3.3-3 of Technical Specification 3.3.2.1, Engineered Safety Features Actuation System Instrumentation, are revised. Changes to the associated nominal setpoint values for the functions contained in the BVPS LRM for Unit 1 and Unit 2 are shown in Attachments C-1 and C-2.

Basis for Change Number 1a

The containment pressure high, containment pressure intermediate high-high and containment pressure high-high values assumed in the revised containment analyses were raised from the current values to provide sufficient margin between the allowable values and the revised containment operating pressure.

The proposed changes to the allowable values are based on the revised safety analysis limits shown in Table 4-3 of Enclosure 2. The calculation of the allowable values are consistent with the BVPS setpoint calculation methodology as described in WCAP-11419, "Westinghouse Setpoint Methodology for Protection Systems, BVPS-Unit 1" (Reference 2) and WCAP-11366, "Westinghouse Setpoint Methodology for Protection Systems, BVPS-Unit 2" (Reference 3). These WCAPs were NRC reviewed and approved as supporting information for Amendments 239 and 120 for Unit 1 and Unit 2 respectively (Reference 4). The approved methodology described in these WCAPs was utilized to determine the new allowable values for each unit.

Change Number 1b

In this change Table 3.3-3, Engineered Safety Feature Actuation System Instrumentation, is revised by changing the allowable values for Refueling Water Storage Tank Level-Low (Unit 1, item 1.1.c) and Refueling Water Storage Tank Level-Extreme Low (Unit 2, item 1.1.b). The BVPS LRM changes to the nominal setpoint values for these functions are shown in Attachment C-1 and C-2.

Basis for Change Number 1b

The Emergency Core Cooling System (ECCS) operates in two phases; the injection phase and the recirculation phase. The injection phase provides borated water from the refueling water storage tank (RWST) to the reactor vessel. The recirculation phase provides long-term post accident cooling by recirculating water from the containment sump. The RWST low level setpoint initiates transfer from the injection to the recirculation phase. The revised RWST level setpoints are based on new analysis requirements, which increased the injected water volume requirements prior to switchover to recirculation from the containment sump. The proposed changes to the allowable values for the RWST transfer to recirculation level setpoints are consistent with the safety analysis limits used in the revised containment analyses and the BVPS trip setpoint methodology in WCAPs 11419 and 11366.

Change Number 1c (Unit 1 change only)

In this change Table 3.3-3, item 1.1.d "Refueling Water Storage Tank Level – Auto QS Flow Reduction" is deleted from the table along with the associated surveillance requirements for this function in Table 4.3-2.

Basis for Change Number 1c (Unit 1 change only)

The current Unit 1 Quench Spray System design includes a restricting flow orifice in parallel with a motor operated cutback valve installed downstream of each quench spray pump to provide quench spray flow for a longer time period to assure containment pressure is maintained sub-atmospheric following a DBA. Upon receipt of a RWST low level signal, the motor operated cutback valve closes and directs water through the flow orifice to provide a reduced flow to the spray nozzles. Since this design feature is not required for an atmospheric containment, it was not modeled in the revised containment analysis. As a result, the Unit 1 automatic quench spray flow cutback on low RWST level will be defeated as part of implementation the approved amendment.

Change Number 2

In this change Technical Specification 3.6.1.3, Containment Air Locks, is revised by changing the value for Pa in Surveillance Requirement 4.6.1.3. Pa is the peak calculated containment internal pressure for the design basis loss of coolant accident (LOCA). A corresponding change is made to Technical Specification 6.17, Containment Leakage Rate Testing Program.

The Unit 1 value for Pa is revised from 40.0 psig to 43.3 psig and the Unit 2 value is revised from 44.7 psig to 44.9 psig.

Basis for Change Number 2

The proposed changes to the values for Pa are consistent with the revised calculated peak accident pressures following a LOCA. See Section 4.7.1 of Enclosure 2 for a detailed discussion on containment pressure response and the revised calculated peak accident pressures following a LOCA. See Section 7.10.1 of Enclosure 2 for a discussion of the impact of the revised Pa values on Technical Specification 6.17, "Containment Leakage Rate Testing Program" and the current measured leakage rates.

Containment leakage rate testing, required by 10CFR50 Appendix J, will continue to be performed in accordance with Technical Specification 6.17, "Containment Leakage Rate Testing Program." The revised Pa value will be incorporated into the program to ensure compliance with Technical Specification 6.17 prior to each unit's first entry into Mode 4 following amendment implementation.

Change Number 3

In this change Technical Specification 3.6.1.4, Internal Pressure, is revised by replacing the air partial pressure requirement with a containment total air pressure requirement of ≥ 12.8 psia and ≤ 14.2 psia. This change eliminates the need for Figure 3.6-1. The word "primary" is also removed from the Limiting Condition for Operations (LCO) statement of this specification.

Basis for Change Number 3

The proposed change, including the deletion of Figure 3.6-1, is consistent with the Improved Standard Technical Specifications (ISTS) LCO for an atmospheric containment design. The revised containment analyses assumptions for initial containment air pressure are ≥ 12.8 psia and ≤ 14.2 psia. The proposed containment total air pressure range will ensure that the containment operating pressure is maintained within these analysis limits. See Section 4.3 of Enclosure 2 for input parameters and assumptions used in the containment analysis. Deletion of the word "primary" from the LCO statement is an administrative change made to conform to the ISTS.

Change Number 4

In this change Technical Specification 3.6.1.5, Air Temperature, for containment average air temperature is revised to replace the existing average air temperature requirement based on Figure 3.6-1, which is deleted as part of Change Number 3, with an average air temperature range of ≥ 70°F and ≤ 105°F. In addition, surveillance requirement (SR) 4.6.1.5 is revised by relocating the specific temperature sensor location details for determining containment average air temperature to the TS Bases. The word "primary" is removed from the LCO statement.

Basis for Change Number 4

This change revises the current minimum air temperature requirements, which were based on satisfying the containment depressurization time analysis for a sub-atmospheric containment. The proposed average air temperature range of ≥ 70°F and ≤ 105°F is consistent with the revised containment analyses assumptions for initial containment temperature. Relocation of the containment air temperature sensor locations from SR 4.6.1.5 to the TS Bases is consistent with the typical location of this information in the ISTS. Reliance on information contained in the TS Bases for guidance in performing this surveillance is acceptable because the TS Bases Control Program specified in the Administrative Controls Section of the TS controls changes to the TS Bases. Deletion of the word "primary" is an administrative change as described above.

Change Number 5 (Unit 1 only)

In this change Technical Specification 3.6.2.3, Chemical Addition System, is revised by deleting surveillance requirement 4.6.2.3.d.4 which verifies that one of the two operating chemical addition pumps in each subsystem will stop following closure of the Unit 1 cutback control valve.

Basis for Change Number 5 (Unit 1 only)

As discussed in proposed Change Number 1c, the Unit 1 automatic quench spray flow reduction feature was not modeled in the revised containment analysis and will be defeated during the implementation phase of the approved amendment. Since the quench spray flow will no longer be reduced on low RWST level, the associated interlock to reduce the number of operating chemical addition pumps upon closure of the quench spray cutback valve will also be defeated. Therefore, surveillance requirement 4.6.2.3.d.4 will no longer be applicable. Refer to Section 6.3.10 of Enclosure 2 for further discussion of this proposed change on the performance requirements of the Chemical Addition System.

Change Number 6

In this change Technical Specification 3.6.3.1, Containment Isolation Valves, is revised to delete surveillance requirements 4.6.3.1.b and 4.6.3.1.e. The remaining surveillance requirements are renumbered due to these deletions.

Deleting these surveillance requirements removes the requirement to cycle each spring or weight loaded containment isolation valve and verify that the valve remains closed with < 1.2 psid differential pressure across the valve and opens when the differential pressure is > 1.2 psid but < 6.0 psid. Associated changes to the containment isolation valve lists in the LRM for Unit 1 and Unit 2 (which currently excludes check valves not subject to Containment Type C Leak Rate Testing from these surveillances) are shown for information in Attachment C-1 and C-2.

Basis for Change Number 6

The basis for surveillance requirements 4.6.3.1.b and 4.6.3.1.e is to ensure that the containment isolation check valves will remain closed preventing the loss of sub-atmospheric pressure in the containment following a DBA. These valves are designed to require, in order to open, a differential pressure across the valve in the normal flow direction exceeding the expected post DBA differential pressure between atmosphere and containment (about 1.2 psi). As a result, leakage into the containment through incoming lines with check valves inside the containment caused by passive failures of such lines between the containment penetration and the outside isolation valve is prevented.

BVPS Amendments 133 and 9 to Unit 1 and Unit 2 (Reference 5) excluded spring and weight loaded containment isolation check valves not subject to containment Type C leak rate testing from the pressure testing requirements of Surveillance Requirement 4.6.3.1. This proposed change would therefore remove the valve cycle and pressure testing requirements for the remaining check valves that are subject to Type C leakage testing. As discussed above, the basis for requiring these pressure tests is to ensure the containment would remain sub-atmospheric in the event of a passive failure of lines between the containment penetration and the outside containment isolation valve. With the proposed change to an atmospheric containment, the plant design basis requirement to return to sub-atmospheric conditions in the containment within one hour following a DBA will no longer be applicable. The basis for this pressure testing would therefore no longer be applicable to

BVPS. The required Type C leak testing and applicable ASME Section XI testing per Technical Specification 4.0.5 and the BVPS Inservice Test Program will continue to be performed on these valves to demonstrate valve operability.

3.0 BACKGROUND

BVPS Units 1 and 2 containments are sub-atmospheric type containments. The containments are maintained at a sub-atmospheric pressure (air partial pressure between 8.9 psia for Unit 1 and 9.0 psia for Unit 2, and 10.5 psia for either unit) during Modes 1 through 4, to limit the peak containment pressure reached during a postulated accident, and to minimize radioactive releases after an accident by returning the containment to sub-atmospheric conditions.

FENOC proposes to modify the containment operating pressure requirements to allow the Unit 1 and 2 containments to operate as atmospheric containments. Specifically, the containment operating pressure requirements will be revised to maintain each containment within a total air pressure range of 12.8 psia to 14.2 psia during Modes 1 through 4.

The requested change to an atmospheric containment would provide several operational benefits including:

(a) reducing the station startup delay associated with establishing a containment vacuum prior to entry into Mode 4;

(b) reducing the potential for personnel injury when entering the containment due to the pressure changes; and

(c) allowing for safer and more frequent containment entries for inspections and monitoring of equipment inside containment.

In addition, the conversion to an atmospheric containment would support FENOC's plan to submit an extended power uprate (EPU) LAR. FENOC is currently in the process of performing the station reviews and analyses for an EPU LAR. The containment conversion also provides margin in the Large Break LOCA calculated peak clad temperature (PCT) needed to support the planned EPU LAR.

Existing Design and Licensing Basis

The current sub-atmospheric containment design analysis is based on the following criteria:

(a) The peak calculated containment atmosphere pressure shall not exceed the design pressure of 45 psig.

(b) The containment shall be depressurized following a design basis accident to below 1 atmosphere absolute pressure in less than 60 minutes.

(c) Once depressurized, the containment shall be maintained at a pressure less than 1 atmosphere absolute for the duration of the accident.

(d) Inadvertent Quench Spray operation shall not result in containment pressure being less than 8 psia.

The containment air partial pressure is maintained as a function of river (service) water temperature in accordance with TS Figure 3.6-1. Limiting the containment air partial pressure and the containment average air temperature per TS 3.6.1.5, ensures that, following a design basis accident, the containment peak pressure would be limited to less than the containment design pressure, and that the containment would depressurize in less than 60 minutes to sub-atmospheric conditions. Controlling containment partial pressure within the established limits also prevents the containment pressure from violating the containment design minimum pressure of 8 psia in the event of an inadvertent actuation of the Quench Spray System.

The limiting DBA considered relative to containment pressure are a main steamline break (MSLB) and a LOCA.

The limiting DBA at Unit 1 for containment peak pressure is a MSLB. The calculated peak containment pressure during this event is 44.2 psig. The calculated peak containment pressure for a Unit 1 LOCA is 40.0 psig. The DBA for the containment depressurization analysis is a pump suction double ended rupture (DER) LOCA with a calculated containment depressurization time of 3,520 seconds. See BVPS Unit 1 Updated Final Safety Analysis Report (UFSAR) Sections 5.6 and 14.3.4.3.

The limiting DBA at Unit 2 for containment peak pressure is a LOCA. The calculated peak containment pressure during this event is 44.7 psig. The calculated peak containment pressure for a Unit 2 MSLB is 40.98 psig. The DBA for the containment depressurization analysis is a pump suction DER

LOCA with a calculated containment depressurization time of 3,510 seconds. See BVPS Unit 2 UFSAR Section 6.2.1.1.

The containment liner design temperature for both units is 280°F. See BVPS Unit 2 UFSAR Section 5.2.2 and Unit 2 UFSAR Section 3.8.1.3. The limiting DBA for peak containment temperature is a MSLB.

The current Unit 1 containment analysis uses mass and energy releases predicted by the LOCTIC code for the blowdown and reflood phases, and the Westinghouse methodologies for the post-reflood phase. The LOCTIC code is also used to calculate the containment parameters such as pressure, temperature, and pump net positive suction head (NPSH), with credit for containment overpressure. The containment is treated as a single node.

The Unit 2 mass and energy releases were developed using a Westinghouse methodology. The LOCTIC code is then used with the Westinghouse provided mass and energies to calculate the containment parameters such as pressure, temperature and pump NPSH (sump level only). The containment is treated as a single node.

The radiological analyses for the LOCA, control rod ejection accident (CREA), and fuel handling accident (FHA) are based upon Alternative Source Term (AST) methodology, i.e., Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 6). The NRC approved the selective implementation of AST for the FHA in Amendments, 241 and 121, which were issued on August 30, 2001 (Reference 7). For the LOCA and the CREA, NRC approval was provided in Amendments, 257 and 139, which were issued on September 10, 2003 (Reference 8). As noted in Section 3.5.3 and 3.5.7 of the NRC SER, the containment leakage assumptions made in the LOCA and CREA dose consequence analyses were intended to support a proposed containment conversion from sub-atmospheric to atmospheric operating conditions. The balance of the radiological accidents is currently based on Technical Information Document (TID) 14844 and NUREG 0800 methodology. These analyses demonstrate compliance with the dose limits of 10CFR100.11 at the site boundary, i.e., the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ), and General Design Criteria 19 for Control Room Habitability.

Proposed Changes to Design and Licensing Basis

The following discussion provides a summary of the design and licensing basis changes included with the containment conversion.

1. The existing containment design analysis criteria is changed to no longer require the containment to depressurize to sub-atmospheric conditions within 60 minutes following a design basis accident and maintain the containment sub-atmospheric for the duration of the accident. The containment design analyses criteria for containment depressurization is revised to require the containment pressure is reduced to less than 50% of the peak calculated pressure for the LOCA within 24 hours after the postulated accident.

2. The new containment integrity analysis and supporting analyses performed for the proposed change to an atmospheric containment include the following design basis methodology changes:

   (a) The Containment Integrity Analysis and Net Positive Suction Head calculations, which are currently based upon the Stone & Webster LOCTIC computer code, are changed to the Modular Accident Analysis Program DBA version (MAAP-DBA) computer code.

   (b) The Unit 1 LOCA Mass and Energy Release calculation methodology, which currently utilizes Stone & Webster LOCTIC, is changed to the NRC approved Westinghouse methodology as described in WCAP-10325-P-A (Reference 9).

   (c) The Main Steamline Break Mass and Energy Release calculation methodology, which currently utilize the Westinghouse MARVEL code, is changed to the LOFTRAN code as described in WCAP-8860-S1-A (Reference 10).

   It should be noted that the radiological dose analyses impacted by the containment conversion are based on the AST methodology and that the control room radiological dose analyses were developed utilizing atmospheric dispersion factors based on ARCON96 methodology. These features are presently incorporated into the BVPS design and licensing basis through the implementation of Amendments 257 and 139 (Reference 8).

3. The proposed change also revises the design basis requirements for various parameters as specified in the BVPS Technical Specifications.

These changes and the basis for the proposed changes are discussed in Section 2 of Enclosure 1.

4. As a result of the revised analysis in support of the containment conversion, the Unit 1 Quench Spray flow reduction scheme will be eliminated. Therefore, approval of the deletion of the Unit 1 Quench Spray flow reduction scheme is being sought with this submittal.

Related Plant Modifications

The following provides a brief description of the plant modifications that will be made to support the conversion to an atmospheric containment. These modifications do not require a LAR, since they will be evaluated under the 10CFR50.59 safety evaluation process. Included with the description of the modification is a statement of the benefit gained by the modification.

Auxiliary Feedwater System Cavitating Venturi

Cavitating venturi flow elements will be installed in each Auxiliary Feedwater System injection line in Unit 1, similar to those currently installed in the Unit 2 Auxiliary Feedwater System. These flow elements are being installed to limit the mass and energy releases into the containment for a MSLB at the proposed EPU level, and to support the mass and energy release analysis performed for the containment conversion.

Rapid Closure of Feedwater Isolation Valves

New rapid closing Feedwater Isolation Valves will be installed in each of the main feedwater lines in Unit 1. These valves are being installed to limit the mass and energy releases into the containment from a MSLB at the proposed EPU level with containment conversion. Changes to the BVPS LRM to include response time requirements for these valves are shown in Attachment C-1.

Installation of a Drainage Port

A drainage port will be installed in the concrete reactor cavity wall of both BVPS units. The drainage port will be installed as described in Section 4.4 of Enclosure 2 and serve to minimize the potential for water holdup in the reactor cavity and enhance sump inventory. The installation of this drainage port does not require any Technical Specification or LRM changes. The installation is evaluated by the 10CFR50.59 safety evaluation process.

4.0 TECHNICAL ANALYSIS

The Beaver Valley Power Station (BVPS) units are currently licensed with containments that are maintained at sub-atmospheric pressure during normal operation. This License Amendment Request (LAR) proposes revising the BVPS Unit 1 and Unit 2 design and licensing basis such that each unit's containment has an atmospheric design. There are three major aspects associated with the revised analysis conducted to support the conversion to an atmospheric containment. These are the determination of new mass and energy releases, a revised containment integrity analyses, and use of the Alternative Source Term methodology. The selective implementation of the Alternative Source Term methodology necessary to support containment conversion has been approved by the NRC and implemented at the BVPS units as Amendments 257 and 139 for Units 1 and 2, respectively. (Reference 8)

It is noted that, although the revised analysis was performed at a bounding proposed EPU core power level of 2900 MWt, this LAR is not requesting approval of the EPU. A separate LAR for an EPU to 2900 MWt is planned for submittal in the future. The revised containment analysis therefore provides conservatism, since the units will be operated at the currently licensed core power level of 2689 MWt until the EPU LAR is approved.

The changes requested by this submittal are based on revised containment analysis utilizing the MAAP-DBA containment analysis code at a bounding proposed EPU core power level of 2900 MWt. The supporting radiological analysis selectively implements the AST methodology as described in Regulatory Guide 1.183 (Reference 6) for dose analysis impacted by the containment conversion.

The methodology described in WCAP-10325-P-A (Reference 9) was used to determine the new mass and energy releases for a LOCA. The methodology, described in WCAP-8860-S1-A (Reference 10), was used to determine the new mass/energy releases for a main steamline break. These mass and energy releases were provided as input to the BVPS specific MAAP-DBA containment analysis code for the LOCA and MSLB analyses.

The revised containment integrity analysis uses the MAAP-DBA containment analysis code. A description of the MAAP-DBA code was provided to the NRC for a pre-application review in FENOC letter L-03-188 (Reference 11). Chapter 9 of Enclosure 2 provides a detailed discussion of the MAAP-DBA code and its methodology. Approval of the application of the MAAP-DBA methodology to BVPS is being sought by this submittal.

The revised containment integrity analysis, discussed in Chapter 4 of Enclosure 2, demonstrates that the containment pressure and temperature responses following any design basis event will remain within the current containment design basis (45 psig for containment structural design pressure and 280°F for containment liner temperature) and that the containment will depressurize to less than 1/2 Pa within 24 hours of a DBA. See Section 4.7.10 of Enclosure 2 for details.

The impact on electrical equipment qualification for the proposed change to an atmospheric containment and the revised post-accident containment temperature and pressure responses have also been evaluated. These evaluations demonstrate that safety related equipment continue to be capable of performing their post accident design basis functions. See Section 7.11 of Enclosure 2 for details.

The revised radiological analyses, which support containment conversion and are discussed in Chapter 5 of Enclosure 2, utilizes the selective application of the AST methodology. The selective application of the AST methodology was originally incorporated into the BVPS design and licensing basis through a revised Fuel Handling Accident Analysis and the implementation of Amendment 241 for Unit 1 and Amendment 121 for Unit 2 (Reference 7). The selective implementation of the AST methodology was expanded to include the LOCA and CREA as part of the control room habitability Technical Specification changes approved by the NRC and implemented by Amendments 257 for Unit 1 and 139 for Unit 2 (Reference 8). It is noted that the dose impact of modifying the control room habitability systems for other design basis events was also assessed and approved by Amendments 257 for Unit 1 and 139 for Unit 2 (Reference 8). The evaluations for the other design basis events demonstrated that the control room doses would remain within the criteria of SRP 6.4 (Reference 12).

The evaluation of the post accident radiological consequences of the containment conversion demonstrates that offsite and control room doses associated with accidents that challenge containment integrity will be within the acceptance criteria of 10CFR50.67 as supplemented by Regulatory Guide 1.183 (Reference 6) and NUREG-0800, Standard Review Plan (SRP) 15.0.1 (Reference 13).

In addition, the impact of long-term post-LOCA containment leakage resulting from containment conversion and the associated selective application of the AST methodology on post-LOCA operator vital access, Emergency Response Facility habitability, equipment qualification integrated dose, and iodine loading associated with post-accident ventilation charcoal filters has been evaluated. These evaluations demonstrate there are no adverse impacts and continued compliance with applicable regulatory criteria is maintained.

A detailed description of these analyses and station evaluations conducted in support of the requested changes is provided in Enclosure 2. The impact of the proposed changes on other safety analyses and plant systems has also been evaluated, as discussed in Chapter 6 of Enclosure 2, and demonstrate acceptable performance.

The proposed changes to the Technical Specifications, allowing operation of both units with an atmospheric containment, are based on the revised containment analyses and supporting evaluations summarized above and discussed in detail in Enclosure 2. These analyses and evaluations demonstrate the safe operation of the units with the conversion to an atmospheric containment.

5.0 REGULATORY SAFETY ANALYSIS

The Beaver Valley Power Station (BVPS) units are currently licensed with a sub-atmospheric containment design that requires the containment to return to sub-atmospheric conditions following a design basis accident (DBA). This License Amendment Request (LAR) proposes revising the BVPS Unit 1 and Unit 2 design and licensing basis such that each unit has an atmospheric containment design. Following issuance of the requested amendments, each unit's containment will be operated at slightly sub-atmospheric conditions, in accordance with the proposed Technical Specification changes, and the containment design and licensing basis will be changed from being sub-atmospheric within 60 minutes following a DBA to containment pressure being less than 50 % of peak calculated containment pressure (Pa) within 24 hours following a DBA. The proposed changes consist of the following.

1) Revision of the allowable values for Containment Pressure-High, Containment Pressure-Intermediate-High-High and Containment Pressure High-High setpoints.

2) Revision of the allowable values for Refueling Water Storage Tank Level setpoints.

3) Deletion of Refueling Water Storage Tank Level – Auto Quench Spray Flow Reduction functional unit (Unit 1 only).

4) Revision of the peak calculated containment pressure (Pa) for a loss of coolant accident (LOCA).

5) Replacement of containment air partial pressure limits with containment total air pressure limits.

6) Revision of the containment average air temperature requirement.

7) Revision of the spring and weight loaded containment isolation valve surveillance requirements.

8) Revision to the Chemical Addition System requirements (Unit 1 only).

In addition to the Technical Specification changes listed above, this submittal also requests NRC approval to change the BVPS design and licensing basis for both units to reflect the following.

1) The Containment Integrity Analysis and Net Positive Suction Head calculations, which are currently based upon the Stone & Webster LOCTIC computer code, are changed to the Modular Accident Analysis Program DBA version (MAAP-DBA) computer code

2) The Unit 1 LOCA Mass and Energy Release calculation methodology, which currently utilizes Stone & Webster LOCTIC, is changed to the NRC approved Westinghouse methodology as described in WCAP-10325-P-A (Proprietary) and WCAP-10326-A (Nonproprietary), "Westinghouse LOCA Mass & Energy Release Model for Containment Design – March 1979 Version," May 1983.

3) The Main Steamline Break Mass and Energy Release calculation methodology, which currently utilizes the Westinghouse MARVEL code, is changed to the LOFTRAN code as described in WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Nonproprietary), "Supplement 1 –

Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986.

The containment analysis supporting this submittal takes credit for a number of plant modifications to be completed before the BVPS units are converted to an atmospheric containment design. One of these modifications is the installation of a drainage port in each unit's reactor cavity wall. The purpose of this modification is to enhance the containment sump inventory during post accident conditions. The other modifications are applicable to Unit 1 only and consist of the installation of cavitating venturis in the auxiliary feedwater system and fast closing isolation valves in the main feedwater system. Installation of the venturis will reduce the mass and energy input to the containment during main steamline break accident conditions. Installation of the fast closing isolation valves also reduces the mass and energy input to the containment by improving the ability to isolate the main feedwater lines in the event of an accident. All of these modifications have undergone the 10CFR50.59 safety evaluation process for both a sub-atmospheric and atmospheric containment design and were determined to not require NRC approval. They are discussed here because the supporting analysis assumes their installation in the plant.

The proposed elimination of the Unit 1 quench spray cutback will also involve equipment modifications that include such items as disabling automatic valve actuation and chemical feed pump trip circuitry. These modifications, however can not be made until the associated proposed Technical Specification is approved.

5.1 No Significant Hazards Consideration

FirstEnergy Nuclear Operating Company (FENOC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10CFR50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No. The Beaver Valley Power Station (BVPS) containments are designed to withstand the internal pressure and temperature resulting from a loss of coolant accident (LOCA), main steamline break (MSLB), feedwater line break, and a control rod ejection accident (CREA). Each of these accidents has been previously analyzed with the results provided in the Updated Final Safety Analysis Report (UFSAR) except the feedwater line break. This accident is not analyzed because the MSLB is more limiting. The affect on containment pressure and temperature due to a CREA is bounded by a LOCA, since a CREA is modeled as a small break LOCA. The probability of occurrence for these accidents is independent of the type of containment. Additionally the supporting plant modifications will not increase the probability of an accident because they perform an accident mitigation function and are not accident initiators. Therefore a change from a sub-atmospheric to an atmospheric containment will not increase the probability of these accidents.

For accident conditions, the proposed changes will potentially impact the reported dose consequences of the LOCA and CREA for both BVPS units. The radiological consequences of these and the remaining design basis accidents are not adversely impacted by the proposed changes because they are within the current BVPS licensing and design basis.

From a containment integrity viewpoint, the limiting DBA presently is the MSLB for Unit 1 and the LOCA for Unit 2. Following the conversion to an atmospheric containment the limiting DBA will be the LOCA for both units. The revised containment integrity analysis demonstrates that with the installation of the supporting plant modifications that the pressures and temperatures associated with the applicable design basis accidents identified above are within the existing containment design limits.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No. The design basis accidents, which could be adversely affected by the proposed changes, have been reanalyzed. These analyses demonstrate that all acceptance criteria have been satisfied. The revised containment integrity analysis demonstrates that the containment will not be subjected to temperatures or pressures that are beyond its design limits. Converting to an atmospheric containment will not result in any new or different kind of accidents because no new accident initiators will be introduced.

The affects of the supporting plant modifications and the proposed Technical Specification changes on plant structures, systems and components (SSC) have been evaluated and it has been verified that the capability of the SSCs to perform their design functions will be retained following approval of the proposed Technical Specification changes and installation of the supporting plant modifications.

Changes to instrumentation setpoints, surveillance requirements, installation of the supporting plant modifications, and the elimination of certain operability requirements will not create the possibility of a new or different type of accident since these changes would not result in significant changes to the manner in which the affected equipment is operated during normal plant operations.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No. The margin of safety attributed to the containment involves both the pressures and temperatures the containment is subjected to following a DBA, and the on-site and offsite dose consequences associated with normal and post DBA operations.

The revised containment analyses demonstrates that, following a DBA; containment peak pressure and temperature will not exceed the containment's design limits and that the containment pressure will not decrease to below 8 psia following the intentional or inadvertent actuation of the quench spray system. Since the containment design limits are not exceeded, the existing margin of safety between these limits and the containment failure limits is not reduced.

Since the current radiological analyses impacted by the containment conversion are conservatively based on atmospheric operation, it is concluded that the existing dose consequence margin of safety will not be impacted when the BVPS units are operated with an atmospheric containment.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FENOC concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10CFR50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

A review of 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants" (Reference 14), was conducted to assess the potential impact associated with the proposed changes. Enclosure 2 contains discussions of the effect of the proposed changes on applicable Regulatory Guides and NUREGs. The following table lists the criterion potentially impacted, and an assessment of the need for a modification to the UFSAR description of BVPS design conformance to the criterion. For purposes of the review performed, the following definitions are presented.

- No = The applicable UFSAR description of BVPS design conformance, or analysis methodology, does not require modification to reflect the proposed changes.
- Yes = The applicable UFSAR description of BVPS design conformance, or analysis methodology, requires modification to reflect the proposed changes.

The paragraphs following the table provide a discussion of the modification to the UFSAR description for each criterion judged as requiring a modification. Although the UFSAR description of BVPS conformance may require a modification, in no case is an exception to the criterion required.

General Design Criteria		Mod
4	Environmental and Missile Design Bases	No
5	Sharing of Structures, Systems, and Components	No
13	Instrumentation and Control	No
16	Containment Design	Yes
17	Electric Power Systems	No
19	Control Room	No
35	Emergency Core Cooling	No
38	Containment Heat Removal	Yes
39	Inspection of Containment Heat Removal System	No
40	Testing of Containment Heat Removal System	No
41	Containment Atmosphere Cleanup	No
44	Cooling Water	No
50	Containment Design Basis	Yes
51	Fracture Prevention of Containment Pressure Boundary	No
52	Capability for Containment Leakage Rate Testing	No
53	Provisions for Containment Testing and Inspection	No
54	Piping Systems Penetrating Containment	No
55	Reactor Coolant Pressure Boundary Penetrating Containment	No
56	Primary Containment Isolation	No
57	Closed System Isolation Valves	No
64	Monitoring Radioactivity Releases	No

5.2.1 Discussion of Impacts

Criterion 16 Containment Design

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment, and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Design Conformance

A steel-lined, reinforced concrete containment structure, maintained at sub-atmospheric pressure, encloses the entire reactor coolant system and provides an essentially leaktight barrier. The containment structure and the engineered safety features are designed to withstand internal and external environmental conditions that may reasonably be expected during the life of the BVPS units and to ensure that the short and long term conditions following a LOCA do not exceed the design bases. Following a DBA, the recirculation and quench spray systems cool and depressurize the containment atmosphere to maintain the pressure and temperature below the containment design values, and return the containment to sub-atmospheric pressure within one hour.

Modification

The design conformance description for Criterion 16 requires replacing sub-atmospheric with atmospheric and to remove the statement that the containment is returned to sub-atmospheric pressure within one hour. With these modifications, conformance to Criterion 16 will be maintained following approval of the proposed changes

Criterion 38 Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to rapidly reduce, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Design Conformance

The containment depressurization/heat removal system conforms to Criterion 38 and consists of the following:

1. Two 100-percent capacity recirculation spray subsystems which remove heat from the containment following a containment isolation phase B (CIB) signal, via the recirculation spray heat exchangers.

2. Two 100-percent capacity quench spray subsystems are also provided to remove heat from the containment atmosphere and provide sodium hydroxide for pH control following a CIB signal.

Each recirculation system and each quench spray subsystem receives power from an independent electrical bus. Each electrical bus is connected to both offsite and onsite power. Leak detection capabilities and containment isolation are discussed in the UFSAR. In accordance with this Criterion, containment isolation valves provide containment isolation at the penetrations.

Modification

The design conformance description for Criterion 38 requires modification to include the statement that containment pressure will be reduced to less than 50% of the peak calculated pressure for the LOCA within 24 hours after the postulated accident.

Criterion 50 Containment Design Basis

The reactor containment structure, including access openings, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and, as required by 10CFR50.44, energy from metal-water and other chemical reactions that may result from the degradation, but not total failure of emergency core cooling from functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Design Conformance

The containment structure, including personnel and equipment hatches, piping and electrical penetrations, and recirculation and quench spray systems, is designed such that the containment structure's design leakage rate is not exceeded under post-LOCA conditions. In addition, the containment structure is designed to withstand, by a sufficient margin, those pressure and temperature conditions resulting from a DBA. This margin reflects all potential energy sources not included in the conservative calculation of peak conditions.

Modification

The design conformance description for Criterion 50 requires modification to reflect the change in the methodology used in determining mass/energy releases following LOCA and containment integrity. These changes will document that the containment design leakage rate will not be exceeded under post-LOCA conditions and that the containment structure is designed to withstand the pressure and temperature conditions resulting from a DBA. Therefore, conformance to Criterion 50 will be maintained following approval of the proposed changes.

5.2.2 Conclusions

In conclusion, based on the considerations discussed above, (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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

Although neither an environmental impact statement nor an environmental assessment is required in connection with the proposed amendment, an evaluation of the impact on the environment was conducted. Chapter 8 of Enclosure 2 provides the details of the evaluation. The evaluation concludes that the proposed amendment:

- will have no impact on the liquid effluent discharges governed by the NPDES Permit,
- will not affect the radioactive liquid and solid waste effluent streams during normal plant operation,
- will result in a small decrease in the radioactive gaseous effluents from the containment during normal plant operation, and
- that accident radiological doses remain within the applicable regulatory dose limits.

Based on this evaluation and the fact that neither an environmental impact statement nor an environmental assessment is required, the proposed amendment will not have an adverse affect on the environment and can thus be deemed acceptable.

7.0 REFERENCES

1. FENOC letter L-03-146, "License Amendment Request Nos. 315 and 188", October 17, 2003.
2. WCAP-11419, Revision 4, March 2004 "Westinghouse Setpoint Methodology for Protection Systems, BVPS-Unit 1".
3. WCAP-11366, Revision 6, March 2004 "Westinghouse Setpoint Methodology for Protection Systems, BVPS-Unit 2".
4. NRC Issuance of Amendment letter dated July 30, 2001, Beaver Valley Power Station License Amendments 239 (Unit 1) and 120 (Unit 2).
5. NRC Issuance of Amendment letter dated December 13, 1988, Beaver Valley Power Station License Amendments 133 (Unit 1) and 9 (Unit 2).
6. Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors".

7. NRC Issuance of Amendment letter dated August 30, 2001, Beaver Valley Power Station License Amendments 241 (Unit 1) and 121 (Unit 2).

8. NRC Issuance of Amendment letter dated September 10, 2003, Beaver Valley Power Station License Amendments 257 (Unit 1) and 139 (Unit 2).

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

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

11. FENOC letter L-03-188, "Containment Conversion Pre-application Report", November 24, 2003.

12. NUREG-0800, Standard Review Plan 6.4, "Control Room Habitability Systems", Revision 2.

13. NUREG-0800, Standard Review Plan 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms", Revision 0.

14. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants."

Attachment A-1

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Changes

License Amendment Request No. 317

The following is a list of the affected pages:

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* No changes are proposed. The page is included for information/readability only.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION AND FEEDWATER ISOLATION						
a. Manual Initiation	2	1	2	Not Applicable	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	Not Applicable	1, 2, 3, 4	13, 36
c. Containment Pressure-High	3	2	2	≤ ~~1.8~~ 5.33 psig	1, 2, 3	14
d. Pressurizer Pressure-Low	3	2	2	≥ 1841 psig	1, 2, 3(1)	14
e. Steamline Pressure-Low	3/loop	2/loop any loop	2/loop any loop	≥ 495.8 psig steam line pressure	1, 2, 3(1)	14

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
1.1 SAFETY INJECTION-TRANSFER FROM INJECTION TO THE RE-CIRCULATION MODE						
a. Manual Initiation	2 sets 2 switches/ set	1 set	2 sets	Not Applicable	1, 2, 3, 4	18
b. Automatic Actuation Logic Coincident with Safety Injection Signal	2	1	2	Not Applicable	1, 2, 3	18
c. Refueling Water Storage Tank Level-Low	4	2	3	≥ ~~18'5"~~ 13′ 9″ and ≤ ~~19'0"~~ 14′ 4″	1, 2, 3	16
~~d. Refueling Water Storage Tank Level - Auto QS Flow Reduction~~	~~1 per train~~	~~1 per train~~	~~1 per train~~	~~≥ 8'2.5" and ≤ 8'9.5"~~	~~1, 2, 3~~	~~18~~

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
2. CONTAINMENT SPRAY						
a. Manual	2 sets	1 set 2 switches	2 sets	Not Applicable	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	Not Applicable	1, 2, 3, 4	13
c. Containment Pressure--High-High	4	2	3	≤ ~~8.3~~ 11.43 psig	1, 2, 3	16
3. CONTAINMENT ISOLATION						
a. Phase "A" Isolation						
1) Manual	2	1	2	Not Applicable	1, 2, 3, 4	18
2) From Safety Injection Automatic Actuation Logic	2	1	2	Not Applicable	1, 2, 3, 4	13
b. Phase "B" Isolation						
1) Manual	2 sets (2 switches/ set)	1 set	2 sets	Not Applicable	1, 2, 3, 4	18
2) Automatic Actuation Logic	2	1	2	Not Applicable	1, 2, 3, 4	13
3) Containment Pressure--High-High	4	2	3	≤ ~~8.3~~ 11.43 psig	1, 2, 3	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION						
a. Manual	2/steam line	1/steam line	2/operating steam line	Not Applicable	1, 2, 3	18
b. Automatic Actuation Logic	2	1	2	Not Applicable	1, 2, 3	13
c. Containment Pressure Intermediate-High-High	3	2	2	≤ ~~3.3~~ 7.33 psig	1, 2, 3	14
d. Steamline Pressure-Low	3/loop	2/loop any loop	2/loop any loop	≥ 495.8 psig steam line pressure	1, 2, 3(1)	14
e. Steamline Pressure Rate-High Negative	3/loop	2/loop any loop	2/operating loop	≤ 104.2 psi with a time constant ≥ 50 seconds	3(2)	14
5. TURBINE TRIP & FEEDWATER ISOLATION						
a. Steam Generator Water Level--High-High, P-14	3/loop	2 loop in any operating loop	2/loop in each operating loop	≤ 81.7% of narrow range instrument span each steam generator	1, 2, 3	14

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.1	SAFETY INJECTION-TRANSFER FROM INJECTION TO THE RECIRCULATION MODE				
	a. Manual Initiation	N.A.	N.A.	R	1, 2, 3, 4
	b. Automatic Actuation Logic Coincident with Safety Injection Signal	N.A.	N.A.	M(1)	1, 2, 3
	c. Refueling Water Storage Tank Level-Low	S	R	M	1, 2, 3
	~~d. Refueling Water Storage Tank Level - Auto QS Flow Reduction~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3~~
2.	CONTAINMENT SPRAY				
	a. Manual Initiation	N.A.	N.A.	R	1, 2, 3, 4
	b. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3, 4
	c. Containment Pressure-High-High	S	R	Q	1, 2, 3

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By performing the following air lock leakage rate testing at the frequency specified in the Containment Leakage Rate Testing Program:

1. Verify no detectable seal leakage when the gap between the door seals is pressurized for at least 2 minutes to:

a) Personnel air lock $\geq P_a$ (~~40.0~~43.3 psig).

b) Emergency air lock $\geq$ 10.0 psig.

or, quantify[7] the air lock door seal leakage to ensure that the leakage rate is $\leq 0.0005\ L_a$ when tested at $\geq P_a$ (~~40.0~~43.3 psig) for the personnel air lock and $\leq 0.0005\ L_a$ when tested at $\geq$ 10.0 psig for the emergency air lock.

2. Conduct the overall air lock leakage tests,[8] at $\geq P_a$ (~~40.0~~43.3 psig), and verify the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$ (~~40.0~~43.3 psig):

a) At the frequency specified in the Containment Leakage Rate Testing Program, and

b) Following maintenance performed on the outer personnel air lock door which may result in a decrease in closure force on any part of the door sealing surface.

b. At least once per 18 months during shutdown by verifying that only one door in each air lock can be opened at a time.

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(7) An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

(8) Results shall be evaluated against the acceptance criteria applicable to LCO 3.6.1.2.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 ~~Primary Containment internal air partial pressure shall be maintained ≥ 8.9 PSIA and within the acceptable operation range (below and to the left of the applicable containment temperature limit line(s)) shown on Figure 3.6-1 as a function of river water temperature.~~ Containment internal air pressure shall be ≥ 12.8 psia and ≤ 14.2 psia.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal air ~~partial~~ pressure ~~< 8.9 PSIA or above the applicable containment temperature limit line(s) shown on Figure 3.6-1,~~ not within the above limits, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The ~~primary~~ containment internal pressure shall be determined to be within the limits at least once per 12 hours.

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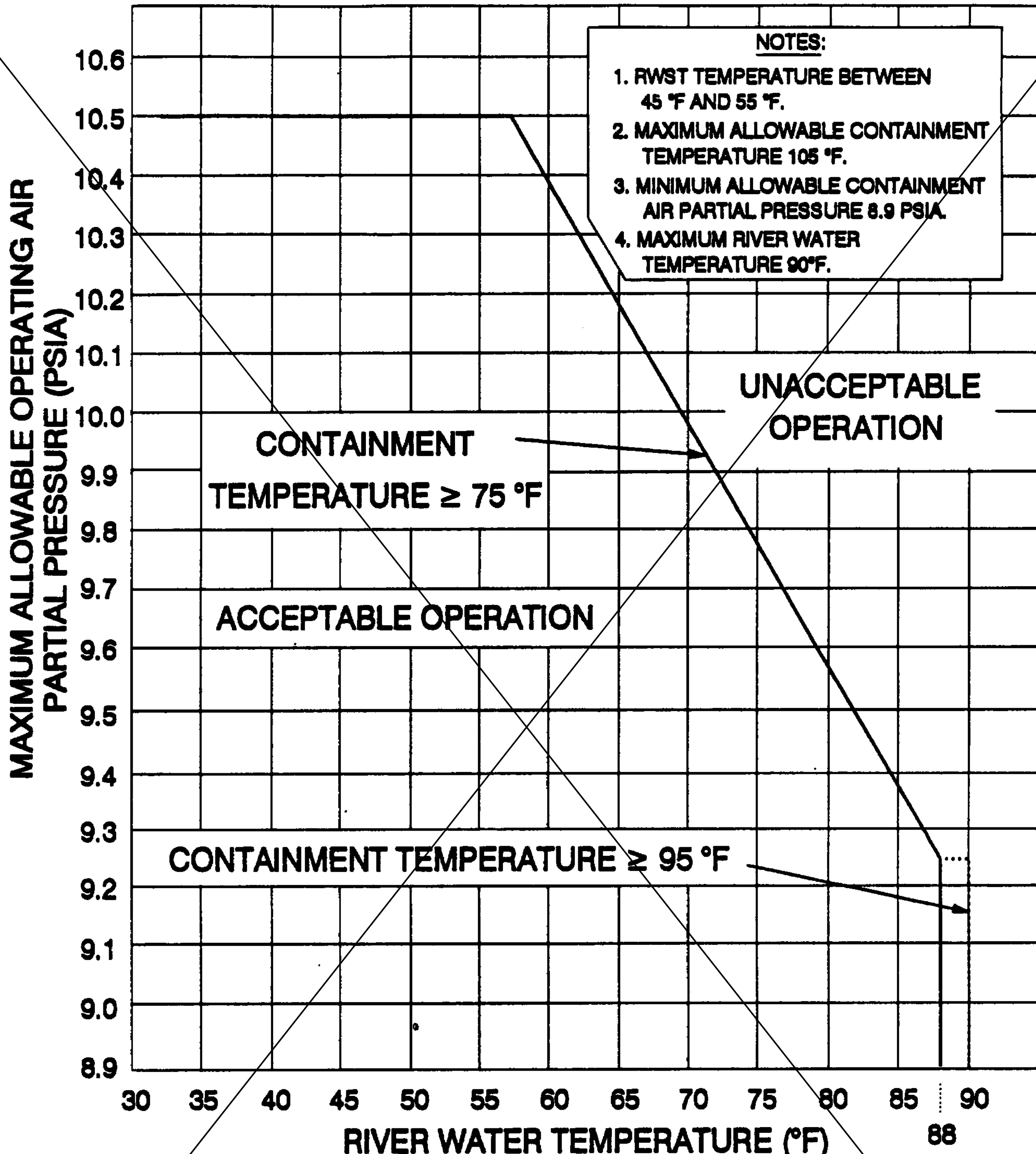

~~FIGURE 3.6-1~~

~~MAXIMUM ALLOWABLE PRIMARY CONTAINMENT AIR PRESSURE VERSUS RIVER WATER TEMPERATURE~~

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 ~~Primaryc~~Containment average air temperature shall be ~~maintained:~~≥ 70°F and ≤ 105°F.

~~a. Greater than or equal to 75°F and less than or equal to 105°F, or~~

~~b. Greater than or equal to 95°F and less than or equal to 105°F~~

~~in accordance with the requirements of Figure 3.6-1.~~

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature ~~> 105°F or less than the minimum containment temperature prescribed in Figure 3.6-1 (75°F or 95°F)~~ not within the above limits restore the average air temperature to within the limits within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 ~~The primary containment average maximum and minimum air temperatures shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours. The nearest alternate detector may be used for temperature determination up to a maximum of one per location.~~The containment average air temperature shall be determined to be within limits at least once per 24 hours.

~~Location~~

~~a. Reactor Containment Annulus - Elev. 701'6"~~

~~b. Reactor Containment Annulus - Elev. 730'0"~~

~~c. Pressurizer Cubicle - Elev. 744'0"~~

~~d. Above Reactor Head Storage Area - Elev. 799'0"~~

~~e. Dome Area above Steam Generator 1B - Elev. 850'0"~~

(next page is 3/4 6-10)

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Figure 3.6-2

~~This Figure has been deleted.~~

~~The technical specification for Initial Average Containment Temperature has been incorporated into 3.6.1.5 (Page 3.6-8) and Figure 3.6-1 (Page 3.6-7).~~

CHEMICAL ADDITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 The chemical addition system shall be OPERABLE with:

a. A chemical addition tank containing at least 4700 gallons of between 19.5 and 20 percent by weight NaOH solution, and

b. Two chemical injection subsystems each capable of adding NaOH solution from the chemical addition tank to a containment quench spray system pump flow.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one chemical addition subsystem inoperable, restore the subsystem to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore the chemical addition subsystem to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The chemical addition system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. By verifying, that on recirculation flow, each pump develops a flow between 25 and 35 gpm when tested pursuant to Specification 4.0.5.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

d. At least once per 18 months, during shutdown, by:

1. Cycling each valve in the chemical addition system flow path that is not testable during plant operation, through at least one complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on a test signal.

3. Verifying that each chemical injection pump starts automatically on a test signal.

~~4. Verifying that in each subsystem with two chemical injection pumps operating, one pump will stop following closure of the cut back control valve.~~

This page contains changes proposed by LAR 315.

3/4.6.3 CONTAINMENT ISOLATION VALVES

This page is included for information/readability only.

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

- - - - - - - - - - - - - - GENERAL NOTES - - - - - - - - - - - - - - - -

1. ACTION a is not applicable to penetration flow paths addressed by ACTION c.

2. ACTION c is only applicable to penetration flow paths with one inoperable containment isolation valve connected to a closed system inside containment.

3. Penetration flow path(s) except for the containment purge supply and exhaust valve flow paths may be unisolated intermittently under administrative controls.

4. Separate ACTION statement entry is allowed for each penetration flow path.

5. Enter applicable ACTION statements for systems made inoperable by containment isolation valves.

- -

a. With one or more penetration flow paths with one containment isolation valve inoperable, isolate the affected penetration flow path within 4 hours by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured; and verify the affected penetration flow path is isolated at least once per 31 days for isolation devices outside containment and prior to entering MODE 4 from MODE 5, if not performed within the previous 92 days, for isolation devices inside containment. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

---

* Locked or sealed closed valves, except for the containment purge supply and exhaust valves, may be opened on an intermittent basis under administrative control.

This page contains changes proposed by LAR 315.

LIMITING CONDITION FOR OPERATION (Continued)

b. With one or more penetration flow paths with two containment isolation valves inoperable, isolate the affected penetration flow path within 1 hour by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one or more penetration flow paths with one containment isolation valve inoperable, isolate the affected penetration flow path within 72 hours by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange; and verify the affected penetration flow path is isolated at least once per 31 days. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE*:

a. By verifying each purge supply and exhaust valve is deactivated in the closed position at least once per 31 days for valves outside containment and prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for valves inside containment.

~~b. By, at least once per 92 days, cycling each weight or spring loaded check valve testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is < 1.2 psid and opens, when the differential pressure in the direction of flow is > 1.2 psid but less than 6.0 psid.~~

~~c~~b. By verifying, at the frequency specified in the Inservice Testing Program, the isolation time of each automatic power operated containment isolation valve that is not locked, sealed, or otherwise secured in position, and required to be closed during accident conditions, is within limits.

* Locked or sealed closed valves, except for the containment purge supply and exhaust valves, may be opened on an intermittent basis under administrative control.

This page contains changes proposed by LAR 315.

SURVEILLANCE REQUIREMENTS (Continued)

~~d~~c. By verifying, at least once per 18 months, each automatic power operated containment isolation valve that is not locked, sealed, or otherwise secured in position, and required to be closed during accident conditions, actuates to the isolation position on an actual or simulated actuation signal.

~~e. By, at least once per 18 months, cycling each weight or spring loaded check valve not testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is < 1.2 psid and opens when the differential pressure in the direction of flow is ≥ 1.2 psid but less than 6.0 psid.~~

OFFSITE DOSE CALCULATION MANUAL (ODCM) (Continued)

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.16 Moved to the PROCESS CONTROL PROGRAM.

6.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions[1]. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, except that the next Type A test performed after the May 29, 1993 Type A test shall be performed no later than May 28, 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is ~~40.0~~43.3 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$ for the overall Type A leakage test and $< 0.60\ L_a$ for the Type B and Type C tests on a minimum pathway leakage rate (MNPLR) basis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ on a maximum pathway leakage rate (MXPLR)[2] basis for Type B and Type C tests and $< 0.75\ L_a$ for Type A tests.

(1) Exemptions to Appendix J of 10 CFR 50 dated November 19, 1984, December 5, 1984, and July 26, 1995.

(2) For penetrations which are isolated by use of a closed valve(s), blind flange(s), or de-activated automatic valve(s), the MXPLR of the isolated penetration is assumed to be the measured leakage through the isolation device(s).

Attachment A-2

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Changes

License Amendment Request No. 190

The following is a list of the affected pages:

3/4 3-16
3/4 3-17
3/4 3-18
3/4 6-5a
3/4 6-6
3/4 6-7
3/4 6-8
3/4 6-15 *
3/4 6-16
3/4 6-17
6-26

* No changes are proposed. The page is included for information/readability only.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION AND FEEDWATER ISOLATION						
a. Manual Initiation	2	1	2	N.A.	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	N.A.	1, 2, 3, 4	13, 36
c. Containment Pressure-High	3	2	2	≤ ~~1.8~~ 5.3 psig	1, 2, 3	14
d. Pressurizer Pressure-Low	3	2	2	≥ 1852 psig	1, 2, 3$^{(1)}$	14
e. Steamline Pressure-Low	3/loop	2/loop any loop	2/operating loop	≥ 494 psig*	1, 2, 3$^{(1)}$	14
1.1 SAFETY INJECTION-TRANSFER FROM INJECTION TO THE RECIRCULATION MODE						
a. Automatic Actuation Logic Coincident with Safety Injection Signal	2	1	2	N.A.	1, 2, 3, 4	18
b. Refueling Water Storage Tank Level-Extreme Low	4	2	3	≥ 31′ 8" and ≤ 31′ 10" ~~37 feet 11 inches~~	1, 2, 3, 4	16

* Time constants utilized in the lead-lag controllers for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
2. CONTAINMENT SPRAY						
a. Manual Initiation	2 sets (2 switches/ set)	1 set	2 sets	N.A.	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	N.A.	1, 2, 3, 4	13
c. Containment Pressure--High-High	4	2	3	≤ ~~8.3~~11.4 psig	1, 2, 3	16
3. CONTAINMENT ISOLATION						
a. Phase "A" Isolation						
1) Manual Initiation	2	1	2	N.A.	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	N.A.	1, 2, 3, 4	13
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.					
b. Phase "B" Isolation						
1) Manual Initiation	2 sets (2 switches/ set)	1 set	2 sets	N.A.	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	N.A.	1, 2, 3, 4	13
3) Containment Pressure--High-High	4	2	3	≤ ~~8.3~~11.4 psig	1, 2, 3, 4	16

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION						
a. Manual Initiation						
1. Individual	1/steam line	1/steam line	1/operating steam line	N.A.	1, 2, 3	41
2. System	2 sets (2 switches/set)	1 set	2 sets	N.A.	1, 2, 3	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	N.A.	1, 2, 3	13
c. Containment Pressure Intermediate-High-High	3	2	2	≤ ~~3.3~~ 7.3 psig	1, 2, 3	14
d. Steamline Pressure-Low	3/loop	2/loop any loop	2/operating loop	≥ 494 psig*	1, 2, 3$^{(1)}$	14
e. Steamline Pressure Rate--High Negative	3/loop	2/loop any loop	2/operating loop	≤ 103.6 psi with a time constant ≥ 50 seconds	3$^{(2)}$	14

* Time constants utilized in the lead-lag controllers for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By performing the following air lock leakage rate testing at the frequency specified in the Containment Leakage Rate Testing Program:

1. Verify no detectable seal leakage when the gap between the door seals is pressurized for at least 2 minutes to:

a) Personnel air lock ≥ Pa (~~44.7~~44.9 psig).

b) Emergency air lock ≥ 10.0 psig.

or, quantify(7) the air lock door seal leakage to ensure that the leakage rate is ≤ 0.0005 L_a when tested at ≥ P_a (~~44.7~~44.9 psig) for the personnel air lock and ≤ 0.0005 L_a when tested at ≥ 10.0 psig for the emergency air lock.

2. Conduct the overall air lock leakage tests,(8) at ≥ P_a (~~44.7~~44.9 psig), and verify the overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a (~~44.7~~44.9 psig):

a) At the frequency specified in the Containment Leakage Rate Testing Program, and

b) Following maintenance performed on the outer personnel air lock door which may result in a decrease in closure force on any part of the door sealing surface.

b. At least once per 18 months during shutdown by verifying that only one door in each air lock can be opened at a time.

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(7) An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

(8) Results shall be evaluated against the acceptance criteria applicable to LCO 3.6.1.2.

LIMITING CONDITION FOR OPERATION

3.6.1.4 ~~Primary Containment internal air partial pressure shall be maintained ≥ 9.0 psia and within the acceptable operation range (below and to the left of the applicable containment temperature limit line) shown on Figure 3.6-1 as a function of service water temperature.~~ Containment internal air pressure shall be ≥ 12.8 psia and ≤ 14.2 psia.

APPLICABILITY: MODES 1, 2, 3 and 4.

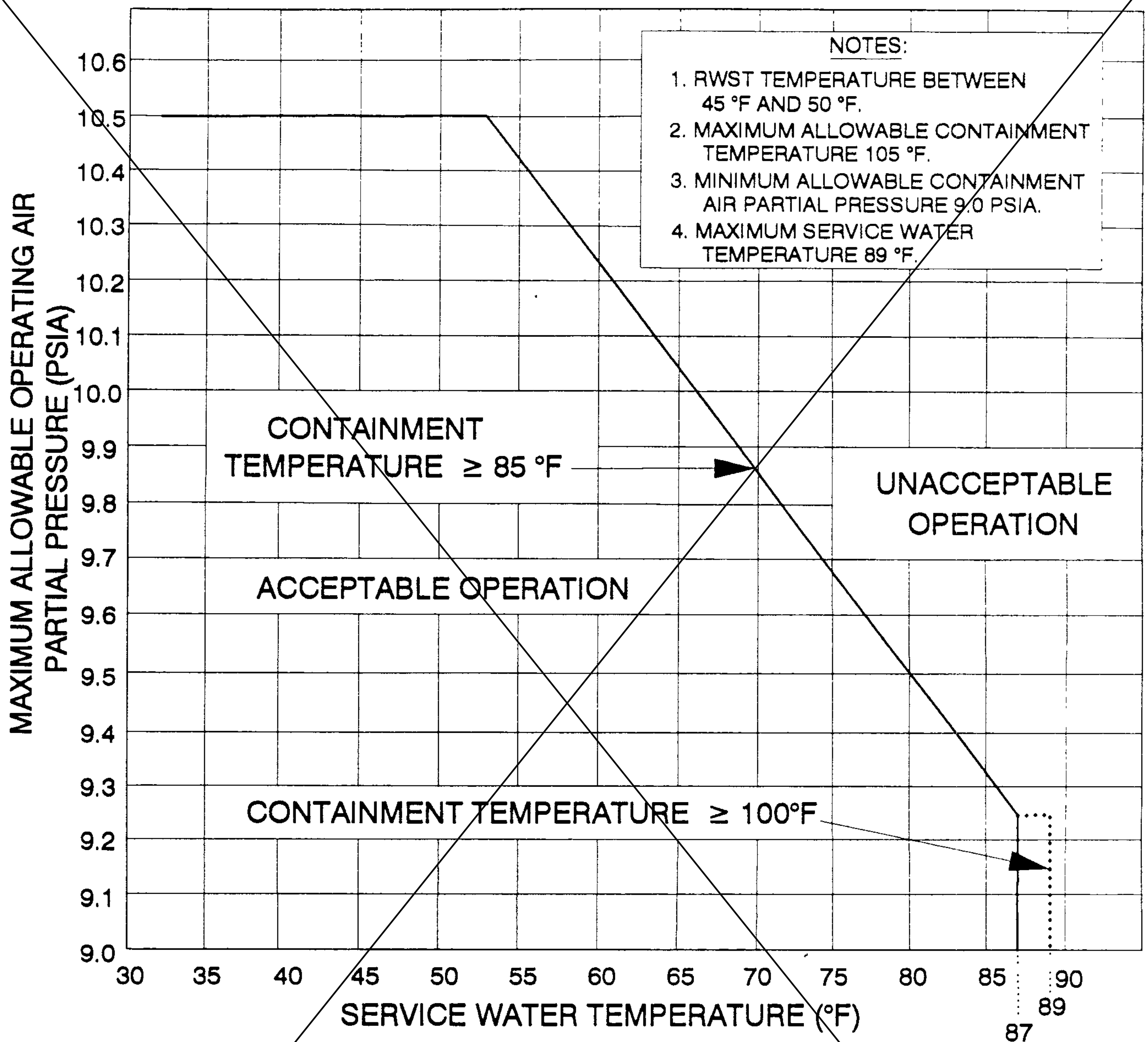
ACTION:

With the containment internal air ~~partial~~ pressure ~~< 9.0 psia or above the applicable containment temperature limit line shown on Figure 3.6-1,~~ not within the above limits, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The ~~primary~~ containment internal pressure shall be determined to be within the limits at least once per 12 hours.

NPF-73



~~FIGURE 3.6-1~~

~~MAXIMUM ALLOWABLE PRIMARY CONTAINMENT AIR PRESSURE~~
~~VERSUS SERVICE WATER TEMPERATURE~~

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AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 ~~Primary c~~Containment average air temperature shall be ~~maintained:~~ ≥ 70°F and ≤ 105°F.

~~a. Greater than or equal to 85°F and less than or equal to 105°F, or~~

~~b. Greater than or equal to 100°F and less than or equal to 105°F~~

~~in accordance with the requirements of Figure 3.6-1.~~

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature ~~> 105°F or less than the minimum containment temperature prescribed in Figure 3.6-1 (85°F or 100°F)~~ not within the above limits restore the average air temperature to within the limits within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 ~~The primary containment average maximum and minimum air temperatures shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours. The nearest alternate detector may be used for temperature determination up to a maximum of one per location.~~ The containment average air temperature shall be determined to be within limits at least once per 24 hours.

~~Location~~

~~a. Reactor Containment Annulus - Elev. 701'6"~~

~~b. Steam Generator 21B Cubicle - Elev. 730'0"~~

~~c. Pressurizer Cubicle - Elev. 743'0"~~

~~d. Reactor Containment Annulus - Elev. 777'4"~~

~~e. Dome area above Steam Generator 21B - Elev. 865'0"~~

This page contains changes proposed by LAR 188.

3/4.6.3 CONTAINMENT ISOLATION VALVES

This page is included for information/readability only.

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE*.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

- - - - - - - - - - - - - - GENERAL NOTES - - - - - - - - - - - - - - - -

1. ACTION a is not applicable to penetration flow paths addressed by ACTION c.

2. ACTION c is only applicable to penetration flow paths with one inoperable containment isolation valve connected to a closed system inside containment.

3. Penetration flow path(s) except for the containment purge supply and exhaust valve flow paths may be unisolated intermittently under administrative controls.

4. Separate ACTION statement entry is allowed for each penetration flow path.

5. Enter applicable ACTION statements for systems made inoperable by containment isolation valves.

- -

a. With one or more penetration flow paths with one containment isolation valve inoperable, isolate the affected penetration flow path within 4 hours by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured; and verify the affected penetration flow path is isolated at least once per 31 days for isolation devices outside containment and prior to entering MODE 4 from MODE 5, if not performed within the previous 92 days, for isolation devices inside containment. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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* Locked or sealed closed valves, except for the containment purge supply and exhaust valves, may be opened on an intermittent basis under administrative control.

This page contains changes proposed by LAR 188.

LIMITING CONDITION FOR OPERATION (Continued)

b. With one or more penetration flow paths with two containment isolation valves inoperable, isolate the affected penetration flow path within 1 hour by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one or more penetration flow paths with one containment isolation valve inoperable, isolate the affected penetration flow path within 72 hours by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange; and verify the affected penetration flow path is isolated at least once per 31 days. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE*:

a. By verifying each purge supply and exhaust valve is deactivated in the closed position at least once per 31 days for valves outside containment and prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for valves inside containment.

~~b. By, at least once per 92 days, cycling each weight or spring loaded check valve testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is < 1.2 psid and opens when the differential pressure in the direction of flow is ≥ 1.2 psid but less than 6.0 psid.~~

b~~c~~. By verifying, at the frequency specified in the Inservice Testing Program, the isolation time of each automatic power operated containment isolation valve that is not locked, sealed, or otherwise secured in position, and required to be closed during accident conditions, is within limits.

* Locked or sealed closed valves, except for the containment purge supply and exhaust valves, may be opened on an intermittent basis under administrative control.

This page contains changes proposed by LAR 188.

SURVEILLANCE REQUIREMENTS (Continued)

c~~d~~. By verifying, at least once per 18 months, each automatic power operated containment isolation valve that is not locked, sealed, or otherwise secured in position, and required to be closed during accident conditions, actuates to the isolation position on an actual or simulated actuation signal.

~~e. By, at least once per 18 months, cycling each weight or spring loaded check valve not testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is < 1.2 psid and opens when the differential pressure in the direction of flow is ≥ 1.2 psid but less than 6.0 psid.~~

CONTAINMENT LEAKAGE RATE TESTING PROGRAM (Continued)

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is ~~44.7~~44.9 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a for the overall Type A leakage test and < 0.60 L_a for the Type B and Type C tests on a minimum pathway leakage rate (MNPLR) basis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a on a maximum pathway leakage rate (MXPLR)(2) basis for Type B and Type C tests and < 0.75 L_a for Type A tests.

b. Air lock testing acceptance criteria and required action are as stated in Specification 3.6.1.3 titled "Containment Air Locks."

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.18 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. a change in the TS incorporated in the license; or

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(2) For penetrations which are isolated by use of a closed valve(s), blind flange(s), or de-activated automatic valve(s), the MXPLR of the isolated penetration is assumed to be the measured leakage through the isolation device(s).

Attachment B-1

Beaver Valley Power Station, Unit No. 1

Proposed Technical Specification Bases Changes

License Amendment Request No. 317

The following is a list of the affected pages:

B 3/4 1-2a
B 3/4 6-2
B 3/4 6-8
B 3/4 6-9
B 3/4 6-10
B 3/4 6-17
B 3/4 6-18

3/4.1.2.8 Refueling Water Storage Tank (RWST)

The minimum required volume of water for the Refueling Water Storage Tank (RWST) provides: 1) a source of water and Net Positive Suction Head (NPSH) for High Head Safety Injection and Low Head Safety Injection (LHSI), 2) adequate sump water for LHSI and Recirculation Spray Pump NPSH, and 3) water for containment Quench Spray. ~~Specifically, the limiting case for defining the minimum RWST volume is derived from the containment analysis for subatmospheric peak pressure during a Reactor Coolant Pump suction Large Break Loss of Coolant Accident.~~ The minimum volume corresponds to 439,050 total gallons as contained in the RWST. From this total volume, the analysis value of 430,500 gallons is considered to be delivered to the respective systems.

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analysis.

The boration capability of the RWST is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirements occur at BOL from full power peak xenon conditions and requires 65,000 gallons of 2400 ppm borated water from the refueling water storage tank.

3/4.1.2.9 Isolation of Unborated Water Sources - Shutdown

Isolation of the primary grade water flow path during MODES 4, 5 and 6 precludes an unplanned boron dilution at these conditions since the sole source of unborated water to the charging pumps is isolated. This eliminates the design basis boron dilution event in MODES 4, 5 and 6. During planned boron dilution events, operator attention will be focused on the boron dilution process and any inappropriate blender operation would be readily identified through various indications which includes the output from the source range nuclear instrumentation.

3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

BACKGROUND (Continued)

extended periods when frequent containment entry is necessary. The emergency air lock, which is located in the equipment hatch opening, is normally removed from the containment building and stored during a refueling outage. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double o-ring seals and local leakage rate testing capability to ensure pressure integrity. DBA conditions which increase containment pressure will result in increased sealing forces on the personnel air lock inner door and both doors on the emergency air lock. The outer door on the personnel air lock is periodically tested in a manner where the containment DBA pressure is attempting to overcome the door sealing forces.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment and containment pressurization are a loss of coolant accident (LOCA) and a control rod ejection accident (CREA). A main steamline break inside containment is not evaluated as the dose consequences are bounded by a main steam line break outside containment. In the analysis of a design basis LOCA or CREA ~~of each of these accidents~~, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1 percent of containment air weight per day. This leakage rate is defined in Specification 6.17 titled "Containment Leakage ~~-~~Rate Testing Program," as L_a = 0.1 percent of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = ~~40.0~~43.3 psig following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

which applies a force which opposes the breech ring force, is not necessary following certain inner air lock door maintenance. SR 4.6.1.3.a.1 sufficiently demonstrates the ability of the inner air lock door to provide a leak tight barrier following maintenance affecting the door sealing surface.

Replacement of the o-rings on the outer personnel air lock door, which results in decreasing the breech ring closure force, will require performance of SR 4.6.1.3.a.2 in addition to SR 4.6.1.3.a.1 which is required due to the door being opened. This surveillance is required because containment DBA pressure tends to overcome the outer personnel air lock door sealing forces. Performance of SR 4.6.1.3.a.1 on the outer personnel air lock applies an opposing force to the breech ring closure force in the same manner as previously described for the inner personnel air lock door. However, for the outer personnel air lock door, the containment pressure developed during a DBA applies an opening force which is opposing the breech ring closure force. Therefore, upon completion of certain maintenance activities, continued outer door leak tightness during a DBA cannot be assured by performance of SR 4.6.1.3.a.1 alone. Maintenance which may result in a decrease in closure force on any part of the door sealing surface (decreasing of breech ring travel for example), will require performance of SR 4.6.1.3.a.2. The performance of this surveillance is necessary to ensure that containment DBA pressure applied against the outer door will not result in the unseating of the air lock door by overcoming of the breech ring closure forces to the point where the leakage becomes excessive. Since SR 4.6.1.3.a.2 duplicates DBA forces on the outer personnel air lock door and also measures the air lock leakage rate, performance of this surveillance requirement demonstrates the continued ability of the outer personnel air lock door to provide a leak tight barrier, during a DBA, following specific maintenance activities.

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY and personnel safety, considering the ~~subatmospheric design~~ containment operating pressure range of 12.8 psia to 14.2 psia, while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur.

3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

The SR has been modified by two Notes. Note (7) states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note (8) has been added to this SR requiring the results to be evaluated against the acceptance criteria applicable to LCO 3.6.1.2. This ensures that air lock leakage is properly accounted for in determining the combined containment leakage rate.

3/4.6.1.4 and 3/4.6.1.5 INTERNAL PRESSURE AND AIR TEMPERATURE

The limitations on containment internal pressure and average air temperature~~ as a function of river water temperature~~ ensure that 1) the containment structure is prevented from exceeding its design negative pressure of 8.0 psia, 2) the containment peak pressure and peak temperature do~~es~~ not exceed the containment design pressure of 45 psig and containment liner design temperature of 280°F during a LOCA or main steamline break~~ conditions~~, and 3)~~ the containment pressure is returned to subatmospheric conditions following a LOCA~~ the environmental conditions inside the containment following a LOCA or main steamline break will be within the containment post accident environmental qualification analysis.

~~The containment internal pressure and temperature limits shown as a function of river water temperature describe the operational envelope that will 1) limit the containment peak pressure to less than its design value of 45 psig and 2) ensure the containment internal pressure returns subatmospheric within 60 minutes following a LOCA.~~

~~The limits on the parameters of Figure 3.6-1 are consistent with the assumptions of the accident analyses.~~

The limits on containment internal pressure and average air temperature are consistent with the assumptions for initial containment conditions used in the containment analysis. The minimum and maximum initial containment pressures assumed in the analysis are 12.8 psia and 14.2 psia. The minimum and maximum initial containment average air temperatures assumed in the analysis are 70°F and 105°F.

Surveillance requirement 4.6.1.5 verifies the containment average air temperature is within the above limits at least once per 24 hours. The containment average air temperature is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The following locations are used to determine the arithmetical average air temperature:

Location

a. Reactor Containment Annulus – Elev. 701’6”
b. Reactor Containment Annulus – Elev. 730’0”
c. Pressurizer cubicle – Elev. 744’0”
d. Above Reactor Head Storage Area – Elev. 799’0”
e. Dome Area above Steam Generator 1B – Elev. 850’0”

The nearest alternate detector may be used for temperature determination up to a maximum of one per location.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of ~~40.0~~43.3 psig in the event of a LOCA. The visual and Type A leakage tests, performed at the frequency specified in the Containment Leakage Rate Testing Program, are sufficient to demonstrate this capability.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT QUENCH AND RECIRCULATION SPRAY SYSTEMS

~~The OPERABILITY of the containment spray systems ensures that containment depressurization and subsequent return to subatmospheric pressure will occur in the event of a LOCA. The pressure reduction and resultant termination of containment leakage are consistent with the assumptions used in the accident analyses.~~The containment spray systems limit post accident pressure and temperature in containment to less than design values and provide a means for long term containment heat removal. Reduction of containment pressure and the iodine removal capacity of the containment spray reduces the release of fission product radioactivity from containment to the environment in event of a DBA to within limits.

The recirculation spray system consists of four 50 percent capacity subsystems each composed of a spray pump, associated heat exchanger and flow path. Two of the recirculation spray pumps and motors are located outside containment (RS-P-2A and RS-P-2B) and two pumps and motors are located inside containment (RS-P-1A and RS-P-1B). The flow path from each pump is piped to an individual 180° recirculation spray header inside containment. Train "A" electrical power and river water is supplied to the subsystems containing recirculation spray pumps RS-P-1A and RS-P-2A. Train "B" electrical power and river water is supplied to the subsystems containing recirculation spray pumps RS-P-1B and RS-P-2B.

Verifying that each quench spray system pump’s developed head at the flow test point is greater than or equal to the required developed head ensures that quench spray system pump performance has not degraded during the cycle. The term “required developed head” refers to the value that is assumed in the Containment Integrity Safety Analysis for the quench spray pump’s developed head at a specific flow point. This value for the required developed head at a flow point is defined as the Minimum Operating Point (MOP) in the Inservice Testing (IST) Program. The verification that the pump’s developed head at the flow test point is greater than or equal to the required developed head is performed by using a MOP curve. The MOP curve is contained in the IST Program and was developed using the required developed head at a specific flow point as a reference point. From the reference point, a curve was drawn which is a constant percentage below the current pump performance curve. Based on the MOP curve, a verification is performed to ensure that the pump’s developed head at the flow test point is greater than or equal to the required developed head. Flow and differential head are normal test parameters of centrifugal pump performance required by Section XI of the ASME Code. Since the quench spray system pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance.

3/4.6.3 CONTAINMENT ISOLATION VALVES (Continued)

to limit offsite doses. Therefore, these valves are required to be deactivated in the closed position during MODES 1, 2, 3, and 4. A containment purge supply or exhaust valve that is deactivated in the closed position must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing control power to the valve operator.

~~SR 4.6.3.1.b~~

~~Check valves that serve a containment isolation function are weight or spring loaded to maintain positive closure when the differential pressure tending to open the check valve is less than 1.2 psid. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 4.6.3.1.b requires verification of the operation of the check valves that are testable during unit operation. The frequency of 92 days is consistent with the Inserice Testing Program requirements for valve testing on a 92 day frequency.~~

SR 4.6.3.1.~~c~~b

Verifying that the isolation time of each automatic power operated containment isolation valve required to be closed during accident conditions is within limits is required to demonstrate OPERABILITY. Automatic power operated containment isolation valves required to be closed during accident conditions close on a Phase A or Phase B containment isolation signal. This surveillance requirement ensures that each power operated containment isolation valve which is required to be closed during accident conditions will isolate in a time period consistent with the assumptions of the safety analyses. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The frequency of this surveillance requirement is in accordance with the Inservice Testing Program.

SR 4.6.3.1.c~~d~~

Automatic power operated containment isolation valves required to be closed during accident conditions close on a Phase A or Phase B containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This surveillance requirement ensures that each automatic power operated containment isolation valve required to be closed during accident conditions will actuate to its isolation position on a Phase A or Phase B containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position

3/4.6.3 CONTAINMENT ISOLATION VALVES (Continued)

under administrative controls. The 18 month frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power. However, this does not preclude performance of this surveillance at power when it can be accomplished in a safe manner. Operating experience has shown that these components usually pass this surveillance when performed at the 18 month frequency. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.

~~SR 4.6.3.1.e~~

~~Check valves that serve a containment isolation function are weight or spring loaded to maintain positive closure when the differential pressure tending to open the check valve is less than 1.2 psid. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 4.6.3.1.e verifies the operation of the check valves that are not testable during unit operation. The frequency of 18 months is based on such factors as the inaccessibility of these valves, the fact that the unit must be shut down to perform the tests, and the successful results of the tests on an 18 month basis during past operation.~~

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA."

The hydrogen recombiner system is designed to maintain the hydrogen concentration in the containment structure below 4 volume percent following a LOCA. The specified system flow rate (50 scfm) is the flow at post LOCA containment conditions (13 psia and 130°F) assumed in the design analysis to assure the hydrogen concentration is maintained below 4 volume percent following a LOCA.

Attachment B-2

Beaver Valley Power Station, Unit No. 2

Proposed Technical Specification Bases Changes

License Amendment Request No. 190

The following is a list of the affected pages:

B 3/4 6-2
B 3/4 6-8
B 3/4 6-9
B 3/4 6-10
B 3/4 6-17
B 3/4 6-18

3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

BACKGROUND (Continued)

opening, is normally removed from the containment building and stored during a refueling outage. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double o-ring seals and local leakage rate testing capability to ensure pressure integrity. DBA conditions which increase containment~~-~~ pressure~~-~~ will result in increased sealing forces on the personnel air lock inner door and both doors on the emergency air lock. The outer door on the personnel air lock is periodically tested in a manner where the containment DBA pressure is attempting to overcome the door sealing forces.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment and containment pressurization are a loss of coolant accident (LOCA) and a control rod ejection accident (CREA). A main steamline break inside containment is not evaluated as the dose consequences are bounded by a main steamline break outside containment. In the analysis of a design basis LOCA or CREA~~ each of these accidents~~, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1 percent of containment air weight per day. This leakage rate is defined in Specification 6.17 titled "Containment Leakage Rate Testing Program," as L_a = 0.1 percent of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = ~~44.7~~44.9 psig following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a

DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

SR 4.6.1.3.a.1 sufficiently demonstrates the ability of the inner air lock door to provide a leak tight barrier following maintenance affecting the door sealing surface.

Replacement of the o-rings on the outer personnel air lock door, which results in decreasing the breech ring closure force, will require performance of SR 4.6.1.3.a.2 in addition to SR 4.6.1.3.a.1 which is required due to the door being opened. This surveillance is required because containment DBA pressure tends to overcome the outer personnel air lock door sealing forces. Performance of SR 4.6.1.3.a.1 on the outer personnel air lock applies an opposing force to the breech ring closure force in the same manner as previously described for the inner personnel air lock door. However, for the outer personnel air lock door, the containment pressure developed during a DBA applies an opening force which is opposing the breech ring closure force. Therefore, upon completion of certain maintenance activities, continued outer door leak tightness during a DBA cannot be assured by performance of SR 4.6.1.3.a.1 alone. Maintenance which may result in a decrease in closure force on any part of the door sealing surface (decreasing of breech ring travel for example), will require performance of SR 4.6.1.3.a.2. The performance of this surveillance is necessary to ensure that containment DBA pressure applied against the outer door will not result in the unseating of the air lock door by overcoming of the breech ring closure forces to the point where the leakage becomes excessive. Since SR 4.6.1.3.a.2 duplicates DBA forces on the outer personnel air lock door and also measures the air lock leakage rate, performance of this surveillance requirement demonstrates the continued ability of the outer personnel air lock door to provide a leak tight barrier, during a DBA, following specific maintenance activities.

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY and personnel safety, considering the ~~subatmospheric design~~ containment operating pressure range of 12.8 psia to 14.2 psia, while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur.

3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

The SR has been modified by two Notes. Note (7) states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note (8) has been added to this SR requiring the results to be evaluated against the acceptance criteria applicable to LCO 3.6.1.2. This ensures that air lock leakage is properly accounted for in determining the combined containment leakage rate.

3/4.6.1.4 and 3/4.6.1.5 INTERNAL PRESSURE AND AIR TEMPERATURE

The limitations on containment internal pressure and average air temperature ~~as a function of service water temperature~~ ensure that 1) the containment structure is prevented from exceeding its design negative pressure of 8.0 psia, 2) the containment peak pressure and peak temperature do~~es~~ not exceed the containment design pressure of 45 psig and containment liner design temperature of 280°F during a LOCA or main steamline break ~~conditions~~, and 3) ~~the containment pressure is returned to subatmospheric conditions following a LOCA~~ the environmental conditions inside the containment following a LOCA or main steamline break will be within the containment post accident environmental qualification analysis.

~~The containment internal pressure and temperature limits shown as a function of service water temperature describe the operational envelope that will 1) limit the containment peak pressure to less than its design value of 45 psig and 2) ensure the containment internal pressure returns subatmospheric within 60 minutes following a LOCA. Additional operating margin is provided if the containment average air temperature is maintained above 100°F as shown on Figure 3.6-1.~~

~~The limits on the parameters of Figure 3.6-1 are consistent with the assumptions of the accident analyses.~~

The limits on containment internal pressure and average air temperature are consistent with the assumptions for initial containment conditions used in the containment analysis. The minimum and maximum initial conditions assumed in the analysis are 12.8 psia and 14.2 psia. The minimum and maximum initial containment average air temperatures assumed in the analysis are 70°F and 105°F.

Surveillance requirement 4.6.1.5 verifies the containment average air temperature is within the above limits once per 24 hours. The containment average air temperature is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The following locations are used to determine the arithmetical average air temperature:

Location

a. Reactor Containment Annulus - Elev. 701'6"
b. Steam Generator 21B Cubicle - Elev. 730'0"
c. Pressurizer Cubicle - Elev. 743'0"
d. Reactor Containment Annulus - Elev. 777'4"
e. Dome area above Steam Generator 21B - Elev. 865'0"

The nearest alternate detector may be used for temperature determination up to a maximum of one per location.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of ~~44.7~~44.9 psig in the event of a LOCA. The visual and Type A leakage tests, performed at the frequency specified in the Containment Leakage Rate Testing Program, are sufficient to demonstrate this capability.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT QUENCH AND RECIRCULATION SPRAY SYSTEMS

~~The OPERABILITY of the containment spray systems ensures that containment depressurization and subsequent return to subatmospheric pressure will occur in the event of a LOCA. The pressure reduction and resultant termination of containment leakage are consistent with the assumptions used in the accident analyses.~~The containment spray systems limit post accident pressure and temperature in containment to less than design values and provide a means for long term containment heat removal. Reduction of containment pressure and the iodine removal capacity of the containment spray reduces the release of fission product radioactivity from containment to the environment in event of a DBA to within limits.

The recirculation spray system consists of four 50 percent capacity subsystems each composed of a spray pump, associated heat exchanger and flow path. All recirculation spray pumps and motors are located outside containment and supply flow to two 360° recirculation spray ring headers located in containment. One spray ring is supplied by the "A" train subsystem containing recirculation spray pump 2RSS-P21A and the "B" train subsystem containing recirculation spray pump 2RSS-P21D with the other spray ring being supplied by the "A" train subsystem containing recirculation spray pump 2RSS-P21C and the "B" train subsystem containing recirculation spray pump 2RSS-P21B. When the water in the refueling water storage tank has reached a predetermined extreme low level, the C and D subsystems are automatically switched to the cold leg recirculation mode of emergency core cooling system operation.

Verifying that each quench spray system pump's developed head at the flow test point is greater than or equal to the required developed head ensures that quench spray system pump performance has not degraded during the cycle. The term "required developed head" refers to the value that is assumed in the Containment Integrity Safety Analysis for the quench spray pump's developed head at a specific flow point. This value for the required developed head at a flow point is defined as the Minimum Operating Point (MOP) in the Inservice Testing (IST) Program. The verification that the pump's developed head at the flow test point is greater than or equal to the required developed head is performed by using a MOP curve. The MOP curve is contained in the IST Program and was developed using the required developed head at a specific flow point as a reference point. From the reference point, a curve was drawn which is a constant percentage below the current pump performance curve. Based on the MOP curve, a verification is performed to ensure that the pump's developed head at the flow test point is greater than or equal to the required developed head. Flow and differential head are normal test parameters of centrifugal pump performance required by

Section XI of the ASME Code. Since the quench spray system pumps cannot be tested with flow through the spray headers, they are tested

3/4.6.3 CONTAINMENT ISOLATION VALVES (Continued)

Surveillance Requirements(SR)

SR 4.6.3.1.a

Each containment purge supply and exhaust valve is required to be verified deactivated in the closed position at least once every 31 days for valves outside containment and prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for valves inside containment. This surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge supply or exhaust valve. The operation of the containment purge supply and exhaust valves has not been evaluated to confirm the ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be deactivated in the closed position during MODES 1, 2, 3, and 4. A containment purge supply or exhaust valve that is deactivated in the closed position must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing control power to the valve operator.

~~SR 4.6.3.1.b~~

~~Check valves that serve a containment isolation function are weight or spring loaded to maintain positive closure when the differential pressure tending to open the check valve is less than 1.2 psid. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 4.6.3.1.b requires verification of the operation of the check valves that are testable during unit operation. The frequency of 92 days is consistent with the Inserice Testing Program requirements for valve testing on a 92 day frequency.~~

SR 4.6.3.1.~~c~~b

Verifying that the isolation time of each automatic power operated containment isolation valve required to be closed during accident conditions is within limits is required to demonstrate OPERABILITY. Automatic power operated containment isolation valves required to be closed during accident conditions close on a Phase A or Phase B containment isolation signal. This surveillance requirement ensures that each power operated containment isolation valve which is required to be closed during accident conditions will isolate in a time period consistent with the assumptions of the safety analyses. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The frequency of this surveillance requirement is in accordance with the Inservice Testing Program.

This page contains changes proposed by LAR 188.

Provided for Information Only.

3/4.6.3 CONTAINMENT ISOLATION VALVES (Continued)

SR 4.6.3.1.c~~d~~

Automatic power operated containment isolation valves required to be closed during accident conditions close on a Phase A or Phase B containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This surveillance requirement ensures that each automatic power operated containment isolation valve required to be closed during accident conditions will actuate to its isolation position on a Phase A or Phase B containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power. However, this does not preclude performance of this surveillance at power when it can be accomplished in a safe manner. Operating experience has shown that these components usually pass this surveillance when performed at the 18 month frequency. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.

~~SR 4.6.3.1.e~~

~~Check valves that serve a containment isolation function are weight or spring loaded to maintain positive closure when the differential pressure tending to open the check valve is less than 1.2 psid. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 4.6.3.1.e verifies the operation of the check valves that are not testable during unit operation. The frequency of 18 months is based on such factors as the inaccessibility of these valves, the fact that the unit must be shut down to perform the tests, and the successful results of the tests on an 18 month basis during past operation.~~

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA."

Attachment C-1

Beaver Valley Power Station, Unit No. 1

Proposed Licensing Requirements Manual Changes

License Amendment Request No. 317

The following is a list of the affected pages:

3.2-2
3.2-3
3.2-4
3.2-5
3.9-4
3.9-5
5.1-3
5.1-4
5.1-15

TABLE 3.2-1
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION			RESPONSE TIME IN SECONDS
1.	Manual		
	a.	Safety Injection (ECCS)	Not Applicable
		Feedwater Isolation	Not Applicable
		Reactor Trip (SI)	Not Applicable
		Containment Isolation-Phase "A"	Not Applicable
		Containment Vent and Purge Isolation	Not Applicable
		Auxiliary Feedwater Pumps	Not Applicable
		Rx Plant River Water System	Not Applicable
	b.	Containment Quench Spray Pumps	Not Applicable
		Containment Quench Spray Valves	Not Applicable
		Containment Isolation-Phase "B"	Not Applicable
	c.	Containment Isolation-Phase "A"	Not Applicable
	d.	Control Room Ventilation Isolation	Not Applicable
2.	Containment Pressure-High		
	a.	Safety Injection (ECCS)	$\leq$ 27.0(3)
	b.	Reactor Trip (from SI)	$\leq$ 3.0
	c.	Feedwater Isolation	
		1) Feedwater Regulating Valves	$\leq$ 10.0(6)
		2) Feedwater Bypass Valves	$\leq$ 30.0(6)
		3) Feedwater Isolation Valves	$\leq$ 10.0(6)
	d.	Containment Isolation-Phase "A"	$\leq$ 22.0(8)/33.0(7)
	e.	Auxiliary Feedwater Pumps	$\leq$ 60.0
	f.	Rx Plant River Water System	$\leq$ 77.0(8)/110.0(7)

TABLE 3.2-1 (Continued)
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27.0[3]/27.0[4]
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	
1) Feedwater Regulating Valves	≤ 10.0[6]
2) Feedwater Bypass Valves	≤ 30.0[6]
3) Feedwater Isolation Valves	≤ 10.0[6]
d. Containment Isolation-Phase "A"	≤ 22.0[8]
e. Auxiliary Feedwater Pumps	≤ 60.0
f. Rx Plant River Water System	≤ 77.0[8]/110.0[7]
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 27.0[4]/37.0[5]
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	
1) Feedwater Regulating Valves	≤ 10.0[6]
2) Feedwater Bypass Valves	≤ 30.0[6]
3) Feedwater Isolation Valves	≤ 10.0[6]
d. Containment Isolation-Phase "A"	≤ 22.0[8]/33.0[7]
e. Auxiliary Feedwater Pumps	≤ 60.0
f. Rx Plant River Water System	≤ 77.0[8]/110.0[7]
g. Steam Line Isolation	≤ 8.0
5. Containment Pressure--High-High	
a. Containment Quench Spray	≤ ~~64.0~~81.5[9]
b. Containment Isolation-Phase "B"	Not Applicable
c. Control Room Ventilation Isolation	≤ 22.0[8]/77.0[7]

TABLE 3.2-1 (Continued)
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip (Above P-9)	Not Applicable
b. Feedwater Isolation	
1) Feedwater Regulating Valves	< 10.0[6]
2) Feedwater Bypass Valves	≤ 30.0[6]
3) Feedwater Isolation Valves	≤ 10.0[6]
7. Containment Pressure--Intermediate High-High	
a. Steam Line Isolation	≤ 8.0
8. Steamline Pressure Rate--High Negative	
a. Steamline Isolation	≤ 8.0
9. Loss of Power	
a. 4.16kv Emergency Bus Undervoltage (Loss of Voltage)	≤ 1.3
b. 4.16kv and 480v Emergency Bus Undervoltage (Degraded voltage)	≤ 95
10. Steam Generator Water Level-Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps[2]	≤ 60.0
b. Turbine-driven Auxiliary Feedwater Pump[1]	≤ 60.0
11. Undervoltage RCP	
a. Turbine-driven Auxiliary Feedwater Pump	≤ 60.0
12. (Deleted)	

TABLE 3.2-1 (Continued)
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
13. Trip of Main Feedwater Pumps	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq$ 60.0

TABLE NOTATION

(1) on 2/3 any Steam Generator

(2) on 2/3 in 2/3 Steam Generators

(3) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps and Low Head Safety Injection pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is not included.

(4) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is included.

(5) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is included.

(6) Feedwater isolation includes signal response and valve closure time.

(7) Diesel generator starting and sequence loading delays included.

(8) Diesel generator starting and sequence loading delays not included.

(9) Diesel generator starting and sequence loading delays included. This response time also includes pump total start time (pump acceleration, begin to deliver flow, etc.) and time to fill the spray piping with water. The maximum allowable isolation valve ~~opening~~ stroke time is ~~considered~~ included in the Quench Spray analysis of record~~, and adequate flow is provided prior to full open of these valves~~. Note that the stroke time of the containment quench spray isolation valves [MOV-1QS-101A, B] is verified in Licensing Requirements Manual Table 5.1-1 "Containment Penetrations."

TABLE 3.9-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT
1. SAFETY INJECTION AND FEEDWATER ISOLATION	
a. Manual Initiation	Not Applicable
b. Automatic Actuation Logic	Not Applicable
c. Containment Pressure-High	~~1.5~~ 5.0 psig
d. Pressurizer Pressure-Low	1845 psig
e. Steamline Pressure-Low	500 psig steam line pressure
1.1 SAFETY INJECTION-TRANSFER FROM INJECTION TO THE RECIRCULATION MODE	
a. Manual Initiation	Not Applicable
b. Automatic Actuation Logic Coincident with Safety Injection Signal	Not Applicable
c. Refueling Water Storage Tank Level-Low	14’ 1/2” ~~18'8-1/2"~~
~~d. Refueling Water Storage Tank Level - Auto QS Flow Reduction~~	~~8'6"~~
2. CONTAINMENT SPRAY	
a. Manual	Not Applicable
b. Automatic Actuation Logic	Not Applicable
c. Containment Pressure--High-High	~~8.0~~ 11.1 psig

TABLE 3.9-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT
3. CONTAINMENT ISOLATION	
a. Phase "A" Isolation	
1. Manual	Not Applicable
2. From Safety Injection Automatic Actuation Logic	Not Applicable
b. Phase "B" Isolation	
1. Manual	Not Applicable
2. Automatic Actuation Logic	Not Applicable
3. Containment Pressure--High-High	~~8.0~~ 11.1 psig
4. STEAM LINE ISOLATION	
a. Manual	Not Applicable
b. Automatic Actuation Logic	Not Applicable
c. Containment Pressure Intermediate-High-High	~~3.0~~ 7.0 psig
d. Steamline Pressure-Low	500 psig steam line pressure
e. Steamline Pressure Rate-High Negative	100 psi with a time constant ≥ 50 seconds
5. TURBINE TRIP & FEEDWATER ISOLATION	
a. Steam Generator Water Level--High-High	81.2% of narrow range instrument span each steam generator

Provided for Information Only.

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENET. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
25	CCR from RCP 1B & 1C Motors	(B)TV-1CC-105D1	< 60	(B)TV-1CC-105D2	< 60
		RV-1CC-272	N/A		
26	CCR from RCP 1A Thermal Barrier	(B)TV-1CC-107E1	< 60	(B)TV-1CC-107E2	< 60
		RV-1CC-273	N/A		
27	CCR from RCP 1A Motor	(B)TV-1CC-105E1	< 60	(B)TV-1CC-105E2	< 60
		RV-1CC-274	N/A		
28	RCS Letdown	(A)TV-1CH-200A	< 60	(A)TV-1CH-204	< 60
		(A)TV-1CH-200B	< 60		
		(A)TV-1CH-200C	< 60		
		(1)MOV-1CH-142	N/A		
		RV-1CH-203	N/A		
29	Primary Drain Transfer Pump #1 Discharge	(A)TV-1DG-108A	< 60	(A)TV-1DG-108B	< 60
		RV-1DG-102	N/A		
30	Spare				
31	Deluge System to Cable Penetration Area	1FP-804	N/A	(A)TV-1FP-105	15
32	Deluge System to RHR Area	1FP-800	N/A	(A)TV-1FP-106	15
33	High Head SI to Hot Legs	(2)(8)1SI-84	N/A	(2)(8)MOV-1SI-869B	N/A
34	Spare				
35	Seal Injection Water RCP 1A	~~(10)~~(2)(8)1CH-181	N/A	(2)(8)MOV-1CH-308A	N/A

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENET. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
36	Seal Injection Water RCP 1B	~~(10)~~(2)(8)1CH-182	N/A	(2)(8)MOV-1CH-308B	N/A
37	Seal Injection Water RCP 1C	~~(10)~~(2)(8)1CH-183	N/A	(2)(8)MOV-1CH-308C	N/A
38	Containment Sump Pump Discharge	(A)TV-1DA-100A RV-1DA-101	< 60 N/A	(A)TV-1DA-100B	< 60
39	Steam Generator 1A Blowdown	Closed System	N/A	(2)(A)TV-1BD-100A	< 60
40	Steam Generator 1B Blowdown	Closed System	N/A	(2)(A)TV-1BD-100B	< 60
41	Steam Generator 1C Blowdown	Closed System	N/A	(2)(A)TV-1BD-100C	< 60
42	Compressed Air to Containment	1SA-15	N/A	1SA-14	N/A
43	Air Activity Monitor Return to Containment	(A)TV-1CV-102-1	< 60	(A)TV-1CV-102	< 60
44	Containment to Air Activity Monitor			(A)TV-1CV-101A (A)TV-1CV-101B	< 60 < 60
45	Primary Grade Water to PRT	1RC-72	N/A	(A)TV-1RC-519	< 60
46	Charging Fill Header	~~(10)~~(2)(8)1CH-170	N/A	(2)(1)(8)FCV-1CH-160	N/A
47	Instrument Air	1IA-91	N/A	1IA-90 (B)TV-1IA-400	N/A < 60
48	Primary Vent Header	(A)TV-1DG-109A2	< 60	(A)TV-1DG-109A1	< 60

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENET. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)

(9) Auto open on Safety Injection recirculation signal.

(10) ~~Not subject to the surveillance requirements of Technical Specification 3/4.6.3. Valves tested per Technical Specification 4.0.5.~~Not used

(11) Valve will be locked shut in Modes 1, 2, 3 and 4.

(12) Valve and associated cap replaced isolation function provided by TV-1LM-100A1 and TV-1LM-100A2.

(13) Isolation is provided by bellows operated hydraulic isolators.

(14) Only one main steam bypass valve is permitted to be open at a time in Modes 1, 2, and 3.

Attachment C-2

Beaver Valley Power Station, Unit No. 2

Proposed Licensing Requirements Manual Changes

License Amendment Request No. 190

The following is a list of the affected pages:

3.2-3
3.2-5
3.10-4
3.10-5
5.1-1
5.1-2
5.1-3
5.1-4
5.1-6
5.1-7
5.1-13
5.1-16

TABLE 3.2-1 (Continued)
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
3.	Pressurizer Pressure-Low	
a.	Safety Injection (ECCS)	≤ 27.0[3]/27.0[4]
b.	Reactor Trip (from SI)	≤ 2.0
c.	Feedwater Isolation	≤ 7.0[6]
d.	Containment Isolation-Phase "A"	≤ 61.0[9]/115.0[10]
e.	Auxiliary Feedwater Pumps	≤ 60.0
f.	Service Water System	≤ 72.0[7]/181.0[8]
4.	Steam Line Pressure-Low	
a.	Safety Injection (ECCS)	≤ 37.0[5]/27.0[4]
b.	Reactor Trip (from SI)	≤ 2.0
c.	Feedwater Isolation	≤ 7.0[6]
d.	Containment Isolation-Phase "A"	≤ 61.0[9]/115.0[10]
e.	Auxiliary Feedwater Pumps	≤ 60.0
f.	Service Water System	≤ 72.0[7]/181.0[8]
g.	Steam Line Isolation	≤ 7.0
5.	Containment Pressure--High-High	
a.	Containment Quench Spray	≤ ~~85.5[10]~~74.5 [12]
b.	Containment Isolation-Phase "B"	Not Applicable
c.	Control Room Ventilation Isolation	≤ 22.0[9]/77.0[10]
6.	Steam Generator Water Level--High High	
a.	Turbine Trip	Not Applicable
b.	Feedwater Isolation	≤ 7.0[6]

TABLE 3.2-1 (Continued)
ENGINEERED SAFETY FEATURES RESPONSE TIMES

TABLE NOTATION

(1) on 2/3 in 2/3 Steam Generators

(2) on 2/3 any Steam Generator

(3) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps and Low Head Safety Injection pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is not included.

(4) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is included.

(5) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is included.

(6) Feedwater system overall response time shall include verification of valve stroke times applicable to the feedwater containment isolation valves for Train A and the main feedwater regulating valves and bypass valves for Train B.

(7) Diesel generator starting and sequence loading delays included. Response time limit includes attainment of discharge pressure for service water pumps.

(8) Diesel generator starting and sequence loading delays not included. Response time limit only includes opening of valves to establish the flowpath to the diesel coolers.

(9) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes operation of valves/dampers.

(10) Diesel generator starting and sequence loading delays included. Response time limit includes operation of valves/dampers.

(11) Diesel generator starting and sequence loading delays not included. Response time limit includes operation of dampers.

(12) Diesel generator starting and sequence loading delays included. Response time does not include operation of the valves because Quench Spray valves are maintained open.

TABLE 3.10-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT			NOMINAL TRIP SETPOINT
1.	SAFETY INJECTION AND FEEDWATER ISOLATION		
	a.	Manual Initiation	N.A.
	b.	Automatic Actuation Logic and Actuation Relays	N.A.
	c.	Containment Pressure - High	~~1.5~~ 5.0 psig
	d.	Pressurizer Pressure - Low	1856 psig
	e.	Steamline Pressure - Low	500 psig*
1.1	SAFETY INJECTION TRANSFER FROM INJECTION TO THE RECIRCULATION MODE		
	a.	Automatic Actuation Logic Coincident with Safety Injection Signal	N.A.
	b.	Refueling Water Storage Tank Level - Extreme Low	~~38~~31 feet ~~0~~9 inches
2.	CONTAINMENT SPRAY		
	a.	Manual Initiation	N.A.
	b.	Automatic Actuation Logic and Actuation Relays	N.A.
	c.	Containment Pressure --High-High	~~8.0~~ 11.1 psig

* Time constants utilized in the lead-lag controllers for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

TABLE 3.10-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT
3. CONTAINMENT ISOLATION	
a. Phase "A" Isolation	
1. Manual Initiation	N.A.
2. Automatic Actuation Logic and Actuation Relays	N.A.
3. Safety Injection	See Functional Unit 1. above for all Safety Injection Trip Setpoints.
b. Phase "B" Isolation	
1. Manual Initiation	N.A.
2. Automatic Actuation Logic and Actuation Relays	N.A.
3. Containment Pressure--High-High	~~8.0~~ 11.1 psig
4. STEAM LINE ISOLATION	
a. Manual Initiation	
1. Individual	N.A.
2. System	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.
c. Containment Pressure Intermediate-High-High	~~3.0~~ 7.0 psig
d. Steam Line Pressure - Low	500 psig*
e. Steamline Pressure Rate -- High Negative	100 psi with a time constant ≥ 50 seconds

* Time constants utilized in the lead-lag controllers for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

TABLE 5.1-1
CONTAINMENT PENETRATIONS

PENT. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
X-1	Comp Cool from Res Heat Exch	(B)2CCP-MOV157-2 2CCP-RV105	< 60 N/A	(B)2CCP-MOV157-1	< 60
X-2	Comp Cool to Res Heat Exch	(B)2CCP-MOV150-2 2CCP-RV102	< 60 N/A	(B)2CCP-MOV150-1	< 60
X-4	Comp Cool to Res Heat Exch	(B)2CCP-MOV151-2 2CCP-RV103	< 60 N/A	(B)2CCP-MOV151-1	< 60
X-5	Comp Cool from Res Heat Exch	(B)2CCP-MOV156-2 2CCP-RV104	< 60 N/A	(B)2CCP-MOV156-1	< 60
X-6	SPARE				
X-7	High Head Safety Injection	(2)~~(13)~~ 2SIS-83	N/A	(2)2SIS-MOV869A	N/A
X-9	SPARE				
X-11	Instrument Air	(A)2IAC-MOV133	< 60	(A)2IAC-MOV134	< 60
X-13	SPARE				
X-14	Chill & Service Wtr to Cont. Air Recirc Cooling Coils	(3)(14)2SWS-MOV153-2	N/A	(3)(14)2SWS-MOV153-1 2SWS-RV153	N/A N/A
X-15	CHARGING	(2)~~(13)~~2CHS-31	N/A	(2)2CHS-MOV289	< 10
X-16	SPARE				

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENT. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
X-17	High Head Safety Injection	(2)~~(13)~~2SIS-84	N/A	(2)2SIS-MOV869B	N/A
X-19	Seal Water from Reactor Coolant Pump	(A)2CHS-MOV378 2CHS-473	< 60 N/A	(A)2CHS-MOV381	< 60
X-20	Safety Injection Accumulator Makeup	2SIS-42	N/A	2SIS-41 2SIS-RV130	N/A N/A
X-21	Chill & Service Wtr from Cont. Air Recirc Cooling Coils	(B)2SWS-MOV155-2	< 60	(B)2SWS-MOV155-1 2SWS-RV155	< 60 N/A
X-22	SPARE				
X-23	SPARE				
X-24	Residual Heat Removal to Refueling Water Tank	2RHS-107	N/A	2RHS-15 2RHS-RV100	N/A N/A
X-25	Chill & Service Wtr from Cont. Air Recirc Cooling Coils	(3)(14)2SWS-MOV154-2	N/A	(3)(14)2SWS-MOV154-1 2SWS-RV154	N/A N/A
X-27	Chill & Service Wtr to Cont. Air Recirc Cooling Coils	(B)2SWS-MOV152-2	< 60	(B)2SWS-MOV152-1 2SWS-RV152	< 60 N/A

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENT. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
X-28	Reactor Coolant Letdown	(A)2CHS-AOV200A	< 60	(A)2CHS-AOV204	< 60
		(A)2CHS-AOV200B	< 60		
		(A)2CHS-AOV200C	< 60		
		(1)2CHS-HCV142	N/A		
		2CHS-RV203	N/A		
X-29	Pri Dr. Trans Pump Disch	(A)2DGS-AOV108A	< 60	(A)2DGS-AOV108B	< 60
				2DGS-RV115	N/A
X-30	SPARE				
X-31	SPARE				
X-32	SPARE				
X-33	SPARE				
X-34	High Head Injection Line	(2)~~(13)~~2SIS-94	N/A	(2)2SIS-MOV836	N/A
				(2)2SIS-MOV840	N/A
X-35	Inj Seal Wtr to Reactor Coolant Pump	(2)~~(13)~~2CHS-474	N/A	(2)2CHS-MOV308A	N/A
X-36	Inj Seal Wtr to Reactor Coolant Pump	(2)~~(13)~~2CHS-476	N/A	(2)2CHS-MOV308B	N/A
X-37	Inj Seal Wtr to Reactor Coolant Pump	(2)~~(13)~~2CHS-475	N/A	(2)2CHS-MOV308C	N/A
X-38	Sump Pump Discharge	(A)2DAS-AOV100A	< 60	(A)2DAS-AOV100B	< 60
				2DAS-RV110	N/A

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENT. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
X-39	St Gen Blowdown	Closed System	N/A	(2)2BDG-AOV100A-1	< 60
X-40	St Gen Blowdown	Closed System	N/A	(2)2BDG-AOV100B-1	< 60
X-41	St Gen Blowdown	Closed System	N/A	(2)2BDG-AOV100C-1	< 60
X-42	Service Air	2SAS-15	N/A	2SAS-14	N/A
X-43	Air Monitor Sample	2CVS-93	N/A	(A)2CVS-SOV102	< 60
X-44	Air Monitor Sample	(A)2CVS-SOV153B	< 60	(A)2CVS-SOV153A	< 60
X-45	Primary Grade Water	2RCS-72	N/A	(A)2RCS-AOV519 2RCS-RV100	< 60 N/A
X-46	Loop Fill	(2)~~(13)~~2CHS-472	N/A	(1)(2)2CHS-FCV160	N/A
X-47	SPARE				
X-48	Primary Vent Header	(A)2VRS-AOV109A-2	< 60	(A)2VRS-AOV109A-1	< 60
X-49	Nitrogen Supply Manifold	2RCS-68	N/A	(A)2RCS-AOV101	< 60
X-50	SPARE				
X-51	SPARE				
X-52	SPARE				
X-53	Nitrogen Manifold	(A)2GNS-AOV101-2	< 10	(A)2GNS-AOV101-1	< 60

Provided for Information Only.

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENT. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
X-60	Low Head Safety Injection Discharge	(2)~~(13)~~2SIS-132	N/A	(2)2SIS-MOV8888B	N/A
X-61	Low Head Safety Injection Discharge	(2)~~(13)~~2SIS-130	N/A	(2)2SIS-MOV8889	N/A
X-62	Low Head Safety Injection Discharge	(2)~~(13)~~2SIS-133	N/A	(2)2SIS-MOV8888A	N/A
X-63	Quench Pump Discharge	2QSS-4	N/A	(B)2QSS-MOV101A 2QSS-RV101A	< 60(4) N/A
X-64	Quench Pump Discharge	2QSS-3	N/A	(B)2QSS-MOV101B 2QSS-RV101B	< 60(4) N/A
X-65	Fuel Transfer Tube	(7)Flange	N/A	(6)2ISC-102	N/A
X-66	Recirc Spray Pump Suction	Open to Containment	N/A	(B)(2)2RSS-MOV155A	< 60(4)
X-67	Recirc Spray Pump Suction	Open to Containment	N/A	(B)(2)2RSS-MOV155C	< 60(4)
X-68	Recirc Spray Pump Suction	Open to Containment	N/A	(B)(2)2RSS-MOV155D	< 60(4)
X-69	Recirc Spray Pump Suction	Open to Containment	N/A	(B)(2)2RSS-MOV155B	< 60(4)
X-70	Recirculation Pump Discharge	(2)~~(13)~~2RSS-29	N/A	(B)(2)2RSS-MOV156A (6) 2RSS-RV156A	< 60(4) N/A

Provided for Information Only.

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENT. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
X-71	Recirculation Pump Discharge	(2)~~(13)~~2RSS-31	N/A	(10)(B)(2)2RSS-MOV156C (6)2RSS-RV156C	< 60(4) N/A
X-73	Main Steam System "A"	Closed System	N/A	(2)2MSS-AOV101A	6
		Closed System	N/A	(2)2MSS-AOV102A	N/A
		Closed System	N/A	(2)(17)2MSS-SOV105A	N/A
		Closed System	N/A	(2)(15)2MSS-SOV120	N/A
		Closed System	N/A	(6)2MSS-SV101A	N/A
		Closed System	N/A	(6)2MSS-SV102A	N/A
		Closed System	N/A	(6)2MSS-SV103A	N/A
		Closed System	N/A	(6)2MSS-SV104A	N/A
		Closed System	N/A	(6)2MSS-SV105A	N/A
	Steam Drains System	Closed System	N/A	(2)2SDS-AOV111A-1	< 60
		Closed System	N/A	(2)2SDS-AOV129B	< 60
	Steam Vent System	Closed System	N/A	(6)2SVS-PCV101A	N/A
		Closed System	N/A	(6)2SVS-HCV104	N/A

Provided for Information Only.

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENT. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
X-110A	SPARE				
X-110B	SPARE				
X-110C	SPARE				
X-110D	SPARE				
X-113	Safety Injection	(2)~~(13)~~2SIS-95	N/A	(2)2SIS-MOV867C (2)2SIS-MOV867D	< 10(4) < 10(4)
X-114	Recirculation Pump Discharge	(2)~~(13)~~2RSS-32	N/A	(10)(B)(2)2RSS-MOV156D (6)2RSS-RV156D	< 60(4) N/A
X-115	Recirculation Pump Discharge	(2)~~(13)~~2RSS-30	N/A	(B)(2)2RSS-MOV156B (6)2RSS-RV156B	< 60(4) N/A
X-116	Fire Protection HVR Filter B	2FPW-388	N/A	(A)2FPW-AOV221	< 60
X-117	Fire Protection HVR Filter A	2FPW-382	N/A	(A)2FPW-AOV204	< 60
X-118A	SPARE				
X-118B	SPARE				
X-118C	SPARE				

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENT. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)

NOTES (Continued):

(6) Not subject to the requirements of Specification 3/4.6.1 and 3/4.6.3. Listed for information only.

(7) Tested under Type "B" testing.

(8) Not used.

(9) Not used.

(10) Auto close on Safety Injection recirculation signal.

(11) Auto open on QSS switchover signal.

(12) Isolation is provided by bellows operated hydraulic isolators.

(13) ~~Not subject to the surveillance requirements of specification 3/4.6.3. Valves tested per specification 4.0.5.~~Not used.

(14) Valve will be locked shut in Modes 1, 2, 3 and 4.

(15) Auto open on Safety Injection Signal.

(16) Valve operability includes remote closure capability.

Attachment D

Beaver Valley Power Station, Unit Nos. 1 and 2

Commitment Summary

License Amendment Request Nos. 317 (Unit 1) and 190 (Unit 2)

Commitment List

The following table identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 in this document. Any other actions discussed in the submittal represent intended or planned actions by FENOC. They are described only as information and are not regulatory commitments. Please notify Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement, at BVPS on (724) 682-5284 of any questions regarding this document or associated regulatory commitments.

COMMITMENT	ENCLOSURE 2 SECTION NO.	DUE DATE
1. Install a drainage port in the Unit 1 concrete reactor cavity wall.	1.1 4.4	Prior to amendment implementation.
2. Install a drainage port in the Unit 2 concrete reactor cavity wall.	1.1 4.4	Prior to amendment implementation.
3. Install Unit 1 Auxiliary Feedwater System cavitating venturi flow elements.	1.1 6.3.16	Prior to amendment implementation.
4. Install new fast closing Unit 1 Main Feedwater Isolation Valves.	1.1 6.3.15	Prior to amendment implementation.

Beaver Valley Power Station Containment Conversion Licensing Report May 2004

FENOC

FirstEnergy Nuclear Operating Company

EXECUTIVE SUMMARY

This report was written to describe the analyses, evaluations performed and the plant modifications to support converting the Beaver Valley Power Station (BVPS-1 and BVPS-2) containment pressure design basis from sub-atmospheric to atmospheric.

Currently both BVPS units are licensed to operate at 2689 MWt reactor core power level with sub-atmospheric containments. The corresponding NSSS power level is 2697 MWt. Stone & Webster LOCTIC methodology and Westinghouse methodology are used to predict mass and energy releases for BVPS-1. Westinghouse methodology is used to predict mass and energy releases for BVPS-2. Both units use the Stone & Webster LOCTIC code to calculate containment pressures and temperatures.

The analyses performed to support operation of both units with atmospheric containments address all Design Bases Accidents (DBA) using NRC approved Westinghouse mass and energy release methodology, as input to the Modular Accident Analysis Program-Design Basis Analysis (MAAP-DBA) code to calculate containment pressures and temperatures. The supporting radiological analysis used "selective" application of the Alternative Source Term (AST) methodology in accordance with 10CFR50.67, the Standard Review Plan 15.0.1, and NRC Regulatory Guide 1.183 to re-evaluate the radiological events impacted by conversion to atmospheric containments (NRC Safety Evaluation Report for Amendments 257/139 dated September 10, 2003).

The supporting analyses are conservatively based on an uprated reactor core power level of 2900 MWt (2910 MWt NSSS Power) in support of future power uprating on both units. The containment analysis demonstrates the containment pressure and temperatures will remain within the current containment design basis of 45 psig and containment liner design basis of 280°F. The radiological analysis also demonstrates the plant will remain within NRC approved limits. After NRC approval of the proposed changes, the units will operate with atmospheric containments at the currently licensed power until a separate Extended Power Uprate (EPU) license amendment request (LAR) is submitted to, and approved by, the Nuclear Regulatory Commission (NRC).

This report describes the analyses, including mass and energy releases, containment pressure and temperature analysis, and radiological analysis, performed to support conversion of the containment pressure design basis from sub-atmospheric to atmospheric. Reviews, conducted to show continued acceptable performance of systems and programs, are also described. Technical Specification changes are identified and justified. The environmental impact evaluation is also provided.

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LIST OF ACRONYMS

AAC	Alternate AC (Power)
ALARA	As Low As Reasonably Achievable
AMSAC	ATWS Mitigation System Activation Circuitry
ANS	American Nuclear Society
AOV	Air Operated Valve
AST	Alternative Source Term
ATWS	Anticipated Transients Without Scram
BELOCA	Best Estimate Loss of Coolant Accident
BFMC	Batelle Frankfurt Model Containment
BVPS	Beaver Valley Power Station
CAR	Containment Atmosphere Recirculation
CC	Containment Conversion
CFR	Code of Federal Regulation
CH/SI	Charging/Safety Injection
CIA	Containment Isolation Phase A
CIB	Containment Isolation Phase B
CRDM	Control Rod Drive Mechanisms
CREA	Control Rod Ejection Accident
CSE	Containment System Experiment
CVCS	Chemical Volume Control System
CVTR	Carolinas and Virginia Test Reactor
DBA	Design Basis Accident
DCH	Direct Containment Heating
DECL	Double Ended Cold Leg
DEHL	Double Ended Hot Leg
DEPS	Double Ended Pump Suction
DER	Double Ended Rupture
DP	Differential Pressure
EAB	Exclusion Area Boundary
EPU	Extended Power Uprate
EQ	Equipment Qualification
ERF	Emergency Response Facility
ER-OL	Environmental Report – Operating License
FAC	Flow Accelerated Corrosion
FAI	Fauske Associates Incorporated
FCV	Flow Control Valve
FES	Final Environmental Statement
FIV	Feedwater Isolation Valve
FRV	Feedwater Regulator Valve
GCM	Generalized Containment Model
GDC	General Design Criteria
HDR	Heiss Dampf Reaktor
HEM	Homogeneous Equilibrium Model
HLSO	Hot Leg Switchover

LIST OF ACRONYMS (Continued)

IBLOCA	Intermediate Break Loss of Coolant Accident
IET	Integral Effects Experiments
IGPM	Imperial Gallon Per Minute
IRS	Inside Recirculation Spray
ISP	International Standard Problem
LAR	License Amendment Request
LERF	Large Early Release Frequency
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
LTCC	Long Term Core Cooling
M&E	Mass and Energy
MAAP-DBA	Modular Accident Analysis Program – Design Basis Analysis
MAO	Maximum Allowable Operating Air
MFLB	Main Feedwater Line Break
MFW	Main Feedwater
MSCV	Main Steam Check Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steamline Break
MWt	Megawatts Thermal
NaOH	Sodium Hydroxide
NPDES	National Pollutant Discharge Elimination System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ORS	Outside Recirculation Spray
OSG	Original Steam Generator
Pa	Calculated Peak Accident Pressure
PAOT	Post Accident Operating Time
PCT	Peak Clad Temperature
PCWG	Power Capability Working Group
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QS	Quench Spray
RCS	Reactor Coolant System
RG	Regulatory Guide
RS	Recirculation Spray
RSG	Replacement Steam Generator
RSS	Recirculation Spray System
RWS	River Water System
RW/SW	River Water/Service Water
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SBO	Station Blackout

LIST OF ACRONYMS (Continued)

SER	Safety Evaluation Report
SG	Steam Generator
SGTP	Steam Generator Tube Plugging
SI	Safety Injection
SIS	Safety Injection System
SW	Service Water
SWS	Service Water System
TDF	Thermal Design Flow
TID	Technical Information Document
TMI	Three-Mile Island
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report

1 INTRODUCTION AND PURPOSE

This report presents the results of work performed to support conversion of Beaver Valley Power Station Unit 1 (BVPS-1) and Beaver Valley Power Station Unit 2 (BVPS-2) from sub-atmospheric to atmospheric containment design.

The Beaver Valley Power Station (BVPS) consists of two (2) Westinghouse three (3) loop pressurized water reactors (PWR). Each reactor is currently licensed to operate at a steady state reactor core power level of 2689 megawatts thermal (MWt). The corresponding Nuclear Steam Supply System (NSSS) power level is 2697 MWt. Each reactor is enclosed within a reinforced concrete steel-lined containment structure. The containments are maintained sub-atmospheric during operation and are returned to sub-atmospheric within one (1) hour after a Design Basis Accident (DBA).

The current BVPS-1 and -2 NSSS power level of 2697 MWt includes a measurement uncertainty recapture power uprate of 1.4% that was implemented in 2001 as part of the BVPS Full Potential Program. The Full Potential Program also includes elements for:

1. Containment Conversion (CC) of the containments from sub-atmospheric to atmospheric design on BVPS-1 and -2

2. Application of Westinghouse best estimate large break loss of coolant accident methodology (BELOCA) to BVPS-1 and -2

3. Implementation of an Extended Power Uprate (EPU) to a reactor core power level of 2900 MWt (2910 MWt NSSS Power) on BVPS-1 and -2

4. Installation of replacement steam generators (RSG) on BVPS-1.

Analyses for these Full Potential Program elements are being performed in an integrated fashion. The License Amendment Request (LAR) for CC will be submitted first. Separate LARs are planned to be submitted in the future for BELOCA and EPU.

To support these future elements of the Full Potential Program, analyses for CC were performed using inputs and assumptions that support both EPU to 2900 MWt and installation of the RSG's on BVPS-1. The CC inputs and assumptions are also consistent with those used in the BELOCA analysis. In this way, analyses performed to support CC such as mass and energy release analysis and containment pressure and temperature response analysis also apply to operation of BVPS-1 and -2 at EPU conditions and to operate BVPS-1 with RSGs.

CC analyses utilize the Modular Accident Analysis Program-Design Basis Analysis (MAAP-DBA) containment analysis code with mass and energy release data generated using Nuclear Regulatory Commission (NRC) approved Westinghouse methodology. The MAAP-DBA methodology is consistent with the current BVPS licensing basis methodology and with that accepted by the NRC for use at other plants. Plant specific approval of the MAAP-DBA code for use at BVPS-1 and -2 is requested in this submittal. The MAAP-DBA code is described in the Beaver Valley Power Station Pre-Application

Report (Reference 1). Plant specific application of MAAP-DBA to BVPS-1 and -2 is described in Chapter 4 of this report.

The analyses demonstrate the calculated containment pressures and temperatures are within the current structural design of the containment. Likewise, the containment subcompartment analysis demonstrate that the structural loading is within the current design. Thus there is no need for structural analysis nor is there a reduction in the structural analysis margins.

An updated site boundary and control room radiological dose analysis was performed for the events impacted by conversion to an atmospheric containment. This analysis utilized selective implementation of the Alternative Source Term (AST) methodology in accordance with 10CFR50.67 “Accident Source Term”, Standard Review Plan 15.0.1 “Radiological Consequences Analysis Using Alternative Source Terms, Rev D, and Regulatory Guide 1.183 “Alternative Radiological Source Terms for Evaluating Design Bases Accidents at Nuclear Power P1ants.” The selective implementation of the AST methodology has been approved by the NRC and implemented at BVPS as Ammendments 257 and 139.

1.1 PROPOSED CHANGES

Proposed methodology discussed in this report include the following:

- Changing the BVPS-1 LOCA mass and energy release calculation method from the Stone & Webster LOCTIC methodology to the Westinghouse methodology in WCAP-10325-P-A (Reference 2).

- Changing the Main Steamline Break (MSLB) mass and energy release calculation methodology to that documented in WCAP-8822 (Reference 3).

- Generating the SBLOCA Mass and Energy’s with MAAP-DBA which is consistent with the approach approved by the NRC at Cook.

- Changing the Containment Integrity and Net Positive Suction Head (NPSH) calculation methodology from LOCTIC to MAAP-DBA.

Changes in parameters currently controlled by the Technical Specifications include:

- Increasing the containment operating pressure range to be 12.8 psia to 14.2 psia total pressure and eliminating the Maximum Allowable Operating (MAO) Air Partial Pressure Curve from the Technical Specifications.

- Decreasing the Refueling Water Storage Tank (RWST) level setpoints for transfer from the safety injection to the recirculation mode.

- Increasing the Containment High Pressure setpoint for Containment Isolation Phase A (CIA).

- Increasing the Containment Intermediate High-High Pressure setpoint.
- Increasing the Containment High-High Pressure setpoint for Containment Isolation Phase B (CIB).
- Changing the calculated Loss of Coolant Accident (LOCA) accident peak containment pressure (P_a) to reflect the analysis results.
- Decreasing the minimum containment temperature operating limit.
- Eliminating the Quench Spray flow reduction (cutback) (BVPS-1 only).
- Revising the Chemical Addition System consistent with elimination of Quench Spray cutback (BVPS-1 only).
- Revising the surveillance requirements for spring and weight loaded containment isolation check valves.

Plant modifications supporting this change include:

- Installation of new fast closing main feedwater isolation valves at BVPS-1.
- Installation of cavitating venturis in the auxiliary feedwater system at BVPS-1.
- Installation of a drainage port in the concrete reactor cavity wall for BVPS-1 and -2 just above the basemat elevation.
- Disabling the Quench Spray cutback feature for BVPS-1.

1.2 BENEFITS OF CHANGES

Conversion to an atmospheric containment will provide margin in the Large Break LOCA calculated Peak Clad Temperature (PCT) to support operating the BVPS units at EPU conditions.

The containment conversion to an atmospheric containment will make containment entry safer by eliminating the need for air packs which allows for safer containment entry and inspections. This change will significantly reduce worker fatigue, temperature stress, increase mobility/agility and visibility as well as enhance communications. Thus, the time required to perform inspections is reduced as low as reasonably achievable (ALARA) and inspections will also be more effective.

The conversion to an atmospheric containment will reduce station startup delay associated with establishing containment vacuum prior to entry into Mode 4.

The new fast acting BVPS-1 main feedwater isolation valves will improve the ability to isolate the BVPS-1 main feedwater lines in the event of an accident. The new BVPS-1 cavitating venturis in the auxiliary feedwater system will reduce the MSLB mass and energy input to the containment during accident conditions.

The proposed methodology changes will incorporate recent technology improvements/methods, including MAAP-DBA containment integrity analysis, Westinghouse mass and energy releases, improved SBLOCA analysis and the water inventory tracking capabilities of MAAP-DBA.

1.3 JUSTIFICATION FOR CHANGES

The above changes are justified based on:

- Unit specific complete reanalysis of containment parameters for all licensing basis cases at an uprated NSSS power (2910 MWt) using MAAP-DBA methodology.

- Evaluation of subcompartment pressurization transients.

- Evaluation of breaks in the BVPS-1 main steam valve room to address the impact of adding cavitating venturis to the auxiliary feedwater lines.

- A review of systems, programs, and testing which determined that the proposed changes are acceptable.

Chapter 2 of this report presents the NSSS parameters used in the analyses performed to support the proposed changes. Detailed discussions identifying the specifics of those analysis/reviews and supporting the Westinghouse mass and energy release methods and MAAP-DBA code are provided in Chapter 3 and Chapter 4. Chapter 5 discusses radiological assessments and Chapter 6 discusses the evaluations and analyses performed to investigate the effects of the proposed changes on other safety analyses, systems, and programs. Chapter 7 discusses the effect of plant programs in support of containment conversion. Chapter 8 discusses a review of the environmental effects of the proposed changes. Chapter 9 provides a description of MAAP-DBA code and the MAAP-DBA methodology used to generate mass & energy releases for both Intermediate Break LOCA (IBLOCA) and Small Break LOCA (SBLOCA).

1.4 EXISTING DESIGN BASES

1.4.1 Containment Analysis Methods

The current design criteria require that:

- Peak pressure associated with a Design Basis Accident (DBA) shall be less than containment design pressure (45 psig).

- The containment shall be depressurized to a sub-atmospheric condition within 60 minutes following a DBA.

- Once depressurized, the containment shall remain sub-atmospheric for the duration of the accident.

- Peak containment liner design basis temperature shall not exceed 280°F.

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

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

The BVPS-1 containment analysis uses mass and energy releases predicted by the LOCTIC code for the blowdown and reflood phases and Westinghouse methodologies for post-reflood phases. The LOCTIC code is also used to calculate the containment parameters such as pressure, temperature, and pump NPSH (with credit for containment overpressure). The containment is treated as a single node.

The BVPS-2 containment analysis uses mass and energy releases developed using Westinghouse methodology. The LOCTIC code is then used with the Westinghouse furnished mass and energies to calculate the containment parameters such as pressure, temperature, and pump NPSH (sump level only with no credit for containment overpressure). The containment is treated as a single node.

The safety related fluid systems supporting the containment are designed to accept a single active failure in the short term, or a single active or passive failure in the long term following any accident. Single active failure is defined as the failure of a powered component to act on demand to perform its design function.

1.4.2 Radiological Design Features

The BVPS-1 and -2 containments are currently maintained at sub-atmospheric pressure during normal operation. Following accidents that result in containment pressurization such as a loss-of-coolant accident (LOCA) or main steam line break accident (MSLB), or a control rod ejection accident (CREA) the containment pressure is reduced to sub-atmospheric within 60 minutes and remains sub-atmospheric for the duration of the event.

From a dose consequence perspective, the MSLB inside containment is not considered a design basis event, as its dose consequences are bounded by the MSLB outside containment.

The radiological analyses for the LOCA, CREA, and fuel handling accident (FHA) are based upon Alternative Source Term (AST) methodology and demonstrate compliance with 10CFR50.67. The NRC approved the selective implementation of AST for the FHA in Licensing Amendments 241 and 121. For the LOCA and the CREA, NRC approval was provided in Licensing Amendments 257 and 139 on September 10, 2003. The balance of the radiological accidents is currently based on Technical Information Document (TID) 14844 and NUREG 0800 methodology. These analyses demonstrate compliance with the dose limits of 10CFR100.11 at the site boundary, i.e., the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) and GDC 19 for Control Room Habitability.

1.5 REVISED DESIGN BASES

1.5.1 Revised Containment Analysis Methods

The revised design criteria require that:

- Peak pressure associated with a DBA shall continue to be less than containment design pressure (45 psig).
- Containment pressure is reduced to less than 50% of the peak calculated pressure (Pa) within 24 hours following a DBA.
- Peak containment liner design basis temperature shall continue to not exceed 280^{o}F.
- Subcompartment walls shall continue to withstand design basis breaks within the subcompartments.
- Inadvertent actuation of containment spray shall continue to not result in a containment pressure less than the minimum containment design pressure of 8 psia.

For both BVPS units, Mass and Energy releases for all accidents, except SBLOCA, are determined using the Westinghouse methodologies of References 2 and 3. The Mass and Energy releases for the SBLOCA are determined using the MAAP-DBA computer code.

The Containment Integrity Analysis uses the MAAP-DBA computer code for both BVPS units.

The NPSH calculations for the Recirculation Spray System pumps and the Low Head Safety Injection pumps (BVPS-1 only) use the MAAP-DBA computer code with some slight differences for each unit. BVPS-1 will continue to credit overpressure and thus rely upon MAAP-DBA to calculate necessary NPSH parameters, i.e., containment pressure, sump temperature, sump level and pump flow and suction loss. BVPS-2 will use a saturated sump model and therefore will use MAAP-DBA to calculate only the sump level and pump flow and suction loss.

The safety related fluid systems supporting the atmospheric containment design continue to be designed to accept a single active failure in the short term, or a single active or passive failure in the long term following any accident. Single active failure remains as defined is Section 1.4.1.

1.5.2 Revised Radiological Design Features

Conversion to an atmospheric containment design will not change the radiological analyses described in Section 1.4.2. Thus following the containment conversion the radiological analysis for LOCA, CREA and FHA will continue to be based upon guidance provided in Regulatory Guide 1.183 and AST methodology, with ARCON96 dispersion methodology and the balance of the radiological accidents will continue to be based upon TID 14844 and NUREG 0800 methodology.

These analyses demonstrate compliance with the dose limits of 10CFR50.67 or 10CFR100.11 and GDC 19, as applicable, at the site boundary, (i.e., the EAB and the LPZ), and Control Room, and assume containment leakage for 30 days following any DBA that results in containment pressurization.

References

1. Beaver Valley Power Station Pre-Application Report dated November 2003 submitted to the NRC by FENOC letter L-03-188 on November 24, 2003.

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

3. WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), “Mass and Energy Releases Following a Steam Line Rupture,” September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-SI-A (Nonproprietary), “Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture,” September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Nonproprietary), “Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Sub-atmospheric Containment Designs,” September 1986.

4. NRC Issuance of Amendment letter dated July 30, 2001, Beaver Valley Power Station License Amendments 239 (Unit 1) and 120 (Unit 2).

5. NRC Issuance of Amendment letter dated December 13, 1988, Beaver Valley Power Station License Amendments 133 (Unit 1) and 9 (Unit 2).

6. NRC Issuance of Amendment letter dated August 30, 2001, Beaver Valley Power Station License Amendments 241 (Unit 1) and 121 (Unit 2).

7. NRC Issuance of Amendment letter dated September 10, 2003, Beaver Valley Power Station License Amendments 257 (Unit 1) and 139 (Unit 2).

2 NSSS PARAMETERS

2.1 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) PARAMETERS

The containment conversion and EPU projects included NSSS performance analyses to develop bounding NSSS Performance Capability Working Group (PCWG) parameters for use in the analyses and evaluations of the NSSS, including NSSS design transients, systems, components, accidents, and nuclear fuel. The resulting NSSS PCWG parameters apply to both projects such that the analyses performed for containment conversion apply to EPU conditions. The NSSS PCWG parameters for BVPS-1 also include parameters for the Model 54F replacement steam generator (RSG) such that the analyses performed for containment conversion on BVPS-1 also apply to EPU conditions with the Model 54F RSGs.

2.1.1 NSSS Design (PCWG) Parameters

2.1.1.1 Introduction

The NSSS design (PCWG) parameters are the fundamental parameters that are used as input in all the NSSS analyses. They provide the Reactor Coolant System (RCS) and secondary system conditions (temperatures, pressures, and flow) that are used as the basis for the NSSS design transients, systems, components, accidents, and fuel analyses and evaluations.

The PCWG parameters are established using conservative assumptions in order to provide bounding conditions appropriate for the NSSS analyses. For example, the assumed RCS flow in generating the primary and secondary side conditions is the Thermal Design Flow (TDF), which is a conservatively low flow that accounts for flow measurement uncertainty and bounds the maximum expected steam generator tube plugging (SGTP) levels.

In order to predict primary and secondary side conditions that bound the way the plant operates, a range of conditions was established for the vessel average temperature (T_{avg}) (i.e., RCS average temperature), feedwater temperature, and the SGTP level. The T_{avg} range was specified between 566.2°F and 580°F, a feedwater temperature range was specified between 400°F and 455°F, and a SGTP level range was specified between 0% and 22%.

The PCWG parameters identified in Table 2.1-1, Table 2.1-2 and Table 2.1-3 provide the PCWG parameters for the current and the EPU power levels. The EPU PCWG parameters bound the operating configurations for the containment conversion project.

2.1.1.2 Input Parameters and Assumptions

Table 2.1-1 shows the PCWG parameters for the current operating conditions.

Table 2.1-2 and Table 2.1-3 show the PCWG parameters which are applicable for containment conversion at the EPU conditions.

The major input parameters and assumptions used in the calculation of the PCWG parameters established for the BVPS-1 and BVPS-2 EPU project are summarized below:

- The power level for the EPU was set at 2900 MWt reactor (2910 MWt NSSS).
- The EPU power level includes a net RCS heat input of 10 MWt.
- The current TDF of 87,200 gpm/loop was maintained.
- Parameters bound operation with both V5H fuel without IFMs and RFA/RFA-2 fuel with IFMs.
- Design core bypass flow is 6.5%.
- A range of SGTP level from 0% to 22% is selected for the analyses.
- A range of full power normal operating T_{avg} from 566.2°F to 580.0°F is selected for the analyses
- A range of full power feedwater temperature of 400°F to 455°F is selected for the analyses.
- Parameters support the existing steam generators for BVPS-1 and BVPS-2 and replacement Model 54F steam generators for BVPS-1.

2.1.1.3 Description of Analyses and Evaluations

Table 2.1-1 provides the PCWG parameters for the current operating conditions.

Tables 2.1-2 and 2.1-3 provide the PCWG parameter cases which were generated and used as the basis for the EPU project.

Table 2.1-2 lists the parameters for BVPS-1 and BVPS-2 and reflects the current steam generators. These parameters incorporate the EPU NSSS power level of 2910 MWt, 0% and 22% SGTP, T_{avg} range from 566.2°F to 580°F, and feedwater temperature range from 400°F to 455°F.

Table 2.1-3 lists the parameters for BVPS-1 and reflects the replacement Model 54F steam generators. These parameters incorporate the EPU NSSS power level of 2910 MWt, 0% and 22% SGTP, T_{avg} range from 566.2°F to 580°F, and a feedwater temperature range from 400°F to 455°F.

These performance capability parameters were used in the containment conversion project and the EPU project. Analyses and evaluations based on the parameter set or sets which were most limiting were performed, so that the analyses would support operation of BVPS-1 and BVPS-2 over the range of conditions specified, including operation of BVPS-1 with the replacement Model 54F steam generators.

2.1.1.4 Acceptance Criteria and Results

The primary acceptance criteria for the determination of the PCWG parameters are that the parameters must provide adequate flexibility and margin for plant operation, while at the same time bound the range of expected operating conditions. These parameters form the basis for the subsequent analyses and evaluations contained in this document. The analyses demonstrate that the plant satisfies all requirements when operated as an atmospheric containment for EPU conditions.

2.1.1.5 Conclusions

The PCWG parameters for the containment conversion project incorporate EPU conditions which bound the current operating conditions.

Table 2.1-1
BVPS-1 and -2 NSSS Design (PCWG) Parameters
Current Operation

THERMAL DESIGN PARAMETERS	Case 1	Case 2
NSSS Power %	101.4	101.4
MWt	2697	2697
10^6 BTU/hr	9203	9203
Reactor Power MWt	2689	2689
10^6 BTU/hr	9175	9175
Thermal Design Flow, loop gpm	87,200	87,200
Reactor 10^6 lb/hr	99.5	99.5
Reactor Coolant Pressure, psia	2250	2250
Core Bypass, %	6.5	6.5
Reactor Coolant Temperature, °F		
Core Outlet	615.1	615.1
Vessel Outlet	610.8	610.8
Core Average	580.3	580.3
Vessel Average	576.2	576.2
Vessel/Core Inlet	541.6	541.6
Steam Generator Outlet	541.3	541.3
Steam Generator		
Steam Temperature, °F	519.0	505.5(1)
Steam Pressure, psia	806	716(1)
Steam Flow, 10^6 lb/hr total	11.81	11.78
Feed Temperature, °F	439.3	439.3
Moisture, % max.	0.25	0.25
Tube Plugging Level (%)	0	30
Zero Load Temperature, °F	547	547
HYDRAULIC DESIGN PARAMETERS		
Mechanical Design Flow, loop gpm	101,400	
Minimum Measured Flow, gpm/total	266,800	

Note:
(1) Steam conditions are limited to minimums of 760 psia and 512.3°F due to component design transient considerations.

Table 2.1-2
BVPS -1 and -2 NSSS Design (PCWG) Parameters
Extended Power Uprate/OSGs

THERMAL DESIGN PARAMETERS	Case 1	Case 2	Case 3	Case 4
NSSS Power %	109.4	109.4	109.4	109.4
MWt	2910	2910	2910	2910
10^6 BTU/hr	9929	9929	9929	9929
Reactor Power MWt	2900	2900	2900	2900
10^6 BTU/hr	9895	9895	9895	9895
Thermal Design Flow, loop gpm	87,200	87,200	87,200	87,200
Reactor 10^6 lb/hr	101.1	101.1	99.3	99.3
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	6.5	6.5	6.5	6.5
Reactor Coolant Temperature, °F				
Core Outlet	608.6	608.6	621.4	621.4
Vessel Outlet	603.9	603.9	617.0	617.0
Core Average	570.6	570.6	584.6	584.6
Vessel Average	566.2	566.2	580.0	580.0
Vessel/Core Inlet	528.5	528.5	543.1	543.1
Steam Generator Outlet	528.2	528.2	542.8	542.8
Steam Generator				
Steam Temperature, °F	503.0[(1)]	493.3[(1)]	518.1	508.5
Steam Pressure, psia	699[(1)]	641[(1)]	799	735
Steam Flow, 10^6 lb/hr total	12.05/13.00	12.03/12.98	12.08/13.04	12.06/13.01
Feed Temperature, °F	400/455	400/455	400/455	400/455
Moisture, % max.	0.25	0.25	0.25	0.25
Tube Plugging Level (%)	0	22	0	22
Zero Load Temperature, °F	547	547	547	547
HYDRAULIC DESIGN PARAMETERS				
Mechanical Design Flow, loop gpm	101,400			
Minimum Measured Flow, gpm/total	266,800			

Note:

(1) Steam conditions are limited to minimums of 700 psia and 503.1°F due to component design transient considerations.

Table 2.1-3
BVPS-1 NSSS Design (PCWG) Parameters
Extended Power Uprate/Model 54F RSGs

THERMAL DESIGN PARAMETERS	Case 1	Case 2	Case 3	Case 4
NSSS Power % MWt 10^6 BTU/hr	109.4 2910 9929	109.4 2910 9929	109.4 2910 9929	109.4 2910 9929
Reactor Power MWt 10^6 BTU/hr	2900 9895	2900 9895	2900 9895	2900 9895
Thermal Design Flow, loop gpm Reactor 10^6 lb/hr	87,200 101.1	87,200 101.1	87,200 99.3	87,200 99.3
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	6.5	6.5	6.5	6.5
Reactor Coolant Temperature, °F Core Outlet Vessel Outlet Core Average Vessel Average Vessel/Core Inlet Steam Generator Outlet	 608.6 603.9 570.6 566.2 528.5 528.2	 608.6 603.9 570.6 566.2 528.5 528.2	 621.4 617.0 584.6 580.0 543.1 542.8	 621.4 617.0 584.6 580.0 543.1 542.8
Steam Generator Steam Temperature, °F Steam Pressure, psia Steam Flow, 10^6 lb/hr total Feed Temperature, °F Moisture, % max. Tube Plugging Level (%)	 500.5(1) 684(1) 12.02/12.97 400/455 0.10 0	 490.2(1) 623(1) 12.01/12.95 400/455 0.10 22	 515.8 783 12.06/13.01 400/455 0.10 0	 505.6 716 12.03/12.98 400/455 0.10 22
Zero Load Temperature, °F	547	547	547	547
HYDRAULIC DESIGN PARAMETERS				
Mechanical Design Flow, loop gpm	101,400			
Minimum Measured Flow, gpm/total	266,800			

Note:
(1) Steam conditions are limited to minimums of 700 psia and 503.1°F due to component design transient considerations.

3 MASS AND ENERGY RELEASE SAFETY ANALYSIS

3.1 LOCA MASS AND ENERGY RELEASES

The uncontrolled release of pressurized high temperature reactor coolant, or loss-of-coolant accident (LOCA), results in release of steam and water into the containment. This, in turn, results in increases in the local subcompartment pressures, and an increase in the global containment pressure and temperature. There are both long and short-term issues relative to a postulated LOCA that must be considered for the Beaver Valley Power Station Units.

The long-term LOCA mass and energy (M&E) releases, addressed in Section 3.1, are utilized as input to the containment analysis. The containment analysis demonstrates that the containment safeguards systems successfully mitigate the consequences of all current licensing basis LOCAs without exceeding the current containment design basis pressure or temperature. Thus, fully maintaining the containment structural margins and rendering structural re-analysis of the containment unnecessary.

The short-term LOCA mass and energy releases, addressed in Section 3.3, are used as input to the subcompartment analysis. The subcompartment analysis addresses the short pressure pulse (generally less than 3 seconds) accompanying a high-energy line pipe rupture within that subcompartment. The subcompartment analysis demonstrates the resultant pressures and temperatures are within the current design basis values for all current licensing basis breaks. Thus, fully maintaining the subcompartment structural margins and rendering structural re-analysis of the subcompartments unnecessary.

3.1.1 Introduction

The current licensing basis LOCA mass and energy release analysis models a core power (including calorimetric error) of 2705 MWt (which is the original Rated Thermal Power (RTP) + error) and Model 51 steam generators for both units. The LOCA mass and energy release analysis that was performed for the Containment Conversion Project models a core power (including calorimetric error) of 2917.4 MWt and bounds both the Model 51 and the planned replacement (Model 54F) steam generators for BVPS-1, and the Model 51M steam generators for BVPS-2.

The current licensing basis analysis for BVPS-1 uses the LOCTIC code (Reference 1) to generate the blowdown phase of the LOCA mass and energy releases and the containment response. Westinghouse methodology is used for the reflood and post-reflood phases. The current BVPS-2 licensing basis LOCA mass and energy release analysis uses the March 1979 Westinghouse LOCA mass and energy release model described in Reference 2. All of these models have been reviewed and approved by the NRC.

The major difference between the Westinghouse LOCA mass and energy release model and the current licensing basis LOCTIC model used on BVPS-1 is that the Westinghouse model calculates the releases separately from the containment response. Therefore, an iteration between the mass and energy release calculation and the containment response calculation is sometimes required to resolve differences in the containment response input assumptions that were made in the mass and energy release calculation with the calculated containment response.

Westinghouse generated the LOCA mass and energy releases for the Containment Conversion Project using the March 1979 LOCA mass and energy release model described in Reference 2. These releases were used in the containment response calculation. Westinghouse also generated additional LOCA mass and energy releases to support the NPSH evaluations. These releases were calculated using the LOCA mass and energy release model described in Reference 5. Both of the Westinghouse LOCA mass and energy release models have been reviewed and approved by the NRC for use on Westinghouse-designed PWRs, including the Beaver Valley Power Station Units.

The Westinghouse generated LOCA mass and energy releases for the first hour were used in the MAAP-DBA containment response analysis. After this time, the break enthalpy is calculated, along with the containment response, by MAAP-DBA.

This section describes the LOCA mass and energy release calculation methodology for the hypothetical double-ended pump suction (DEPS) and double-ended hot-leg (DEHL) break cases. It also explains that the analysis of the DEPS and DEHL LOCAs bounds all current licensing basis LOCAs, including the double-ended cold leg (DECL) break.

3.1.2 Input Parameters and Assumptions

The mass and energy release analysis is sensitive to the characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the Reactor Coolant System (RCS) operating temperatures are chosen to bound the highest average coolant temperature range of all operating cases, and a temperature uncertainty allowance of (+4.0°F on BVPS-1 and +4.0°F on BVPS-2) is then added. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2250 psia plus an uncertainty allowance (+40 psi on BVPS-1 and +42 psi on BVPS-2).

All input parameters are chosen consistent with accepted analysis methodology. Some of the most critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed next. Tables 3.1-1, 3.1-2 and 3.1-3 for BVPS-1 and Tables 3.1-28, 3.1-29 and 3.1-30 for BVPS-2 present key data assumed in the analysis.

The core rated power adjusted for calorimetric error (+0.6 percent of power) was used in the analysis. As previously noted, the use of RCS operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures that are at the maximum levels attained in steady state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures. The selection of 2290 psia on BVPS-1 and 2292 psia on BVPS-2 as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term mass and energy release calculations.

The selection of the fuel design features for the long-term mass and energy release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (i.e., to maximize the core stored energy). The uncertainty included in core-stored energy modeled for BVPS-1 is +15 percent. For BVPS-2, the margin in core-stored energy is a statistical value that is dependant upon fuel type, power level, and burn up. Thus, the analysis very conservatively accounts for the stored energy in the core.

The nominal RCS volume is increased by 3 percent (1.6 percent allowance for thermal expansion and 1.4 percent for uncertainty) for the LOCA mass and energy release calculation. This assumption helps maximize the initial RCS mass and energy.

A uniform steam generator tube plugging (SGTP) level of zero percent (0%) is modeled. This assumption maximizes the reactor coolant volume and fluid release by considering the RCS fluid in all SG tubes. During the post-blowdown period the steam generators are active heat sources, as significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The zero percent SGTP assumption maximizes heat transfer area and therefore, the transfer of secondary heat across the steam generator (SG) tubes. Additionally, this assumption reduces the reactor coolant loop resistance, which reduces the pressure drop upstream of the break for the pump suction breaks and increases break flow. Thus, the analysis very conservatively accounts for the level of SGTP.

The initial steam generator fluid mass is calculated at full power, and then increased by 10% to cover uncertainties. In order to conservatively represent and bound both steam generators types for the BVPS-1 analysis, conservative steam generator water and metal masses were used. The steam generator energy content bounds both the Model 51 and the planned Model 54F steam generators. For BVPS-2, steam generator water mass and metal mass used conservatively bounds the Model 51 steam generator.

Portions of the SG secondary metal, such as the upper elliptical head, upper shell, and miscellaneous upper internals, have poor heat transfer due to their location in the steam region. The mass of this metal is approximately 216,300 lbm per SG for the Model 51 steam generator and 189,000 lbm per SG for the Model 54 steam generator. The stored energy in this metal will be transferred to the RCS and released to the containment at a much slower rate and is not considered during the first hour of the LOCA mass and energy release calculation for the double-ended pump suction breaks. The stored energy in the rest of the SG secondary metal and fluid is released to the containment within the first hour.

After one hour, the Westinghouse LOCA mass and energy calculation has extracted all of the stored energy from the RCS, except for the stored metal energy in the steam generator upper internals and upper elliptical heads. This energy is assumed to be removed at a constant rate over the next six hours and is added to the core decay heat as an energy source for the long-term steaming rate calculation.

Regarding safety injection flow, the mass and energy release calculation considered configurations/failures to conservatively bound respective alignments. These cases include (1) a

Minimum Safeguards case (one Charging/Safety Injection pump (CH/SI) and one Low Head Safety Injection (LHSI) pump) and (2) a Maximum Safeguards case (two CH/SI and two LHSI pumps).

In summary, the following assumptions were employed so that the LOCA mass and energy releases are conservatively calculated, thereby maximizing the energy release to containment:

1. The nominal RCS volume is increased by 3 percent (1.6-percent allowance for thermal expansion and 1.4 percent for uncertainty)

2. The reactor is assumed to be operating at full core rated power (2900 MWt) plus an allowance for a calorimetric error of +0.6 percent of power.

3. The full power core-stored energy (above T-avg) is increased by 15 percent to account for fuel manufacturing tolerances and an additional allowance is included to account for fuel densification (BVPS-1).

   Core-stored energy is based on the time in life for maximum fuel densification. The assumptions used to calculate the fuel temperatures for the core-stored energy calculation account for appropriate uncertainties associated with the models in the PAD code (e.g., calibration of the thermal model, pellet densification model, cladding creep model, etc.) (Ref. 7). In addition, the fuel temperatures for the core-stored energy calculation account for appropriate uncertainties associated with manufacturing tolerances (e.g., pellet as-built density). The total uncertainty for the fuel temperature calculation is a statistical combination of these effects and is dependent upon fuel type, power level, and burn up (BVPS-2)

4. The RCS is assumed to be at the maximum expected full power operating temperature and an allowance for temperature measurement uncertainty (+4.0°F on BVPS-1 and +4.0°F on BVPS-2) is added. These uncertainties conservatively include both deadband and bias.

5. The RCS is assumed to be at the nominal RCS pressure and an allowance for pressure measurement uncertainty (+40 psi on BVPS-1 and +42 psi on BVPS-2) is added.

6. Conservatively high heat transfer coefficients (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer) are modeled. The SG secondary stored energy is released in one hour. All of the additional stored energy in the upper elliptical head, upper shell, and miscellaneous upper internals, is released at a constant rate over the next 6 hours.

7. The LOCA back-pressure is assumed to remain at the containment design pressure (45 psig). This assumption determines the end of the blowdown phase and minimizes the safety injection flow rate during the reflood phase.

8. A uniform SGTP level of 0% is assumed. This assumption:

   - Maximizes reactor coolant volume and fluid release,

- Maximizes heat transfer area across the SG tubes,

- Reduces coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow.

9. The full power SG level is used to calculate the initial secondary mass and 10% is added to cover uncertainty.

10. The initial secondary energy inventory bounds both the current and planned replacement SGs for BVPS-1.

11. Main feedwater addition following a signal to close the flow control valve.

Thus, based on the previously discussed conditions and assumptions, a separate but bounding analysis for each of the Beaver Valley Power Station Units was made for the release of mass and energy releases from the RCS in the event of a LOCA at the future uprated conditions.

3.1.3 Description of Analyses

The evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in Reference 2. This evaluation model has been reviewed and approved generically by the Nuclear Regulatory Commission (NRC). The approval letter is included with Reference 2.

3.1.3.1 LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA mass and energy release analysis, is typically divided into four phases.

Blowdown – the period of time from accident initiation (when the reactor is at steady state operation) to the time that the RCS and containment reach an equilibrium state.

Refill – the period of time when the lower plenum is being filled by accumulator and Emergency Core Cooling System (ECCS) water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy releases to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

Reflood – begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

Post-reflood (Froth) – describes the period following the reflood phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators prior to exiting the break as steam. After the broken loop steam generator cools, the break flow becomes two-phase.

3.1.3.2 Computer Codes

The Reference 2 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the LOCA mass and energy releases for the Beaver Valley Power Station Containment Conversion Project.

SATAN VI calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, mass and energy flow rates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the ECCS refills the reactor vessel and provides cooling to the core. The most important feature of WREFLOOD is the steam/water mixing model, discussed in subsection 3.1.5.2.

FROTH models the post-reflood portion of the transient. The FROTH code calculates the heat release from the energy stored in the secondary fluid and metal masses, excluding the upper internals and upper elliptical head. This part of the steam generator metal mass is not actively cooled by the two-phase fluid circulating through steam generator tubes and takes longer to cooldown.

EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to containment design pressure to the end of the transient (1 hour). It also compiles a summary of data on the entire transient, including formal instantaneous mass and energy release tables and mass and energy releases balance tables with data at critical times.

For the pump suction break after one hour, the Westinghouse LOCA mass and energy releases calculation has extracted all of the stored energy from the RCS, except for the stored metal energy in the steam generator upper internals and upper elliptical heads. This energy is assumed to be removed at a constant rate over the next six hours and is added to the core decay heat as an energy source for the long-term steaming rate calculation. See Section 3.1.5.6.

3.1.3.3 Break Size and Location

Generic studies (Reference 2, Chapter 3) have been performed to determine the effect of postulated break size on the LOCA mass and energy releases. The double-ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and post-reflood phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture for any release purposes:

- Hot leg (between vessel and steam generator)
- Cold leg (between pump and vessel)
- Pump suction (between steam generator and pump)

The DEHL break location yields the highest blowdown mass and energy release rates (Reference 2, Section 3.3). Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary side is minimal because the majority of fluid that exits the core vents directly to containment, bypassing the steam generators. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction, or cold-leg break locations where the core exit mixture must pass through the steam generators before venting through the break. Studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease) for the hot leg break. Therefore, the mass and energy releases for the blowdown phase of the hot-leg break are calculated and used in the containment peak pressure and temperature response calculation.

Studies have determined that the blowdown transient for the DECL break is, in general, less limiting than that for the pump suction break (Reference 2, Section 3.3). The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. The flooding rate during the reflood phase is greatly reduced, and the energy release rate into the containment is reduced. Therefore, the cold-leg break is bounded by other breaks and no further evaluation is necessary.

The pump suction break combines the effects of the relatively high core-flooding rate, as in the hot-leg break, and the additional stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment.

Therefore, the break locations that were analyzed for this program were the DEPS rupture (10.48 ft^2) and the DEHL rupture (9.2 ft^2). LOCA mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown phase with this methodology. See Section 3.1.6 for long term DEHL cases developed for pump NPSH calculations.

3.1.3.4 Application of Single-Failure Criterion

The mass and energy release calculation assumes a complete loss of all offsite power coincident with the LOCA. The emergency diesel generators are actuated to provide power for the safety injection system. The combination of signal delay plus diesel delay and additional delays in starting the ECCS pumps results in the delivery of SI after the end of blowdown.

For BVPS-1 and -2, two cases are analyzed to assess the effects of a single failure in the mass and energy release calculation. The first case assumes a single failure of one of the emergency diesel generators, resulting in the loss of one train of safeguards equipment. This, in combination with other conservative assumptions (maximum resistances, minimum pump head-flow curves), minimizes the safety injection flow rate. The second case assumes a failure in the containment spray system. The safety injection flow rate for this case is maximized by assuming both trains of safeguards equipment are operating and by including other conservative assumptions (minimum resistances, maximum pump head-flow curves). In addition to these two cases, BVPS-2 also investigates a third case that assumes a failure of one service water (SW) train.

3.1.4 Acceptance Criteria

A large LOCA is classified as an American Nuclear Society (ANS) Condition IV event, an infrequent fault. To satisfy the NRC acceptance criteria presented in the Standard Review Plan (SRP) Section 6.2.1.3, the relevant requirements are as follows:

- 10 CFR 50, Appendix A
- 10 CFR 50, Appendix K, paragraph I.A

To meet these requirements, the following must be addressed:

- Sources of energy
- Break size and location
- Calculation of each phase of the accident

3.1.5 Results

3.1.5.1 Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform, and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in Reference 2.

Table 3.1-4 (BVPS-1) and Table 3.1-31 (BVPS-2) present the calculated mass and energy release for the blowdown phase of the DEHL break. For the hot-leg break mass and energy release tables, break path 1 refers to the mass and energy releases exiting from the reactor vessel side of the break; and break path 2 refers to the mass and energy releases exiting from the steam generator side of the break.

Table 3.1-5 (BVPS-1) and Table 3.1-32 (BVPS-2) present the calculated mass and energy releases for the blowdown phase of the DEPS break with either minimum or maximum ECCS flows. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy releases exiting from the steam generator side of the break; break path 2 refers to the mass and energy releases exiting from the pump side of the break.

3.1.5.2 Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models—one for the contents of the reactor vessel and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena, such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator releases are included as auxiliary equations that interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break, the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the (Reference 2) mass and energy release evaluation model in recent analyses, for example, D.C. Cook (Reference 3). Even though the Reference 2 model credits steam/water mixing only in the intact loop and not in the broken loop, the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 3). Moreover, this assumption is supported by test data and is further discussed below.

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that needs to be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the REFLOOD steam/water mixing model. This data, generated in 1/3-scale tests (Reference 4), are the largest scale data available and thus, most clearly simulate the flow regimes and gravitational effects that would occur in a Pressurized Water Reactor (PWR). These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

A group of 1/3-scale tests corresponds directly to the reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 2. For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the REFLOOD calculation is therefore wholly supported by the 1/3-scale steam/water mixing data. Descriptions of the test and test results are contained in References 2 and 3.

The calculated DEPS reflood phase LOCA mass and energy releases are given in Table 3.1-6 (BVPS-1) and Table 3.1-33 (BVPS-2) for the minimum safeguards case and in Table 3.1-9 (BVPS-1) and Table 3.1-36 (BVPS-2) for the maximum safeguards case and Table 3.1-45 (BVPS-2) for the service water failure case. The transient responses of the principal parameters during reflood are given in

Table 3.1-7 (BVPS-1) and Table 3.1-34 (BVPS-2) for the DEPS minimum safeguards case and in Table 3.1-10 (BVPS-1) and Table 3.1-37 (BVPS-2) for the DEPS maximum safeguards case and Table 3.1-46 (BVPS-2) for the service water failure case.

3.1.5.3 Post-Reflood Mass and Energy Release Data

The FROTH code (References 2 and 5) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken-loop and intact-loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs, and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two-phase. During the FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The FROTH code calculation stops when the secondary side equilibrates to the saturation temperature (T_{sat}) at the containment design pressure. After this point, the EPITOME code completes the SG depressurization. The methodology for the use of this model is described in Reference 2. (See subsection 3.1.5.5 and subsection 3.1.5.6 for additional information.)

Table 3.1-8 (BVPS-1) and Table 3.1-35 (BVPS-2) present the two-phase post-reflood mass and energy release data for the double-ended pump suction case minimum safeguards case. Table 3.1-11 (BVPS-1) and Table 3.1-38 (BVPS-2) present the two-phase post-reflood mass and energy release data for the double-ended pump suction maximum safeguards case. Table 3.1-47 (BVPS-2) presents the release data for the double-ended pump suction service water failure case.

3.1.5.4 Decay Heat Power Model

The American Nuclear Society Standard ANSI/ANS-5.1-1979 (Reference 6) has been used for the determination of decay heat in the mass and energy release analysis. Table 3.1-12, Part 1 lists the generic decay heat curve used in the Beaver Valley Power Station mass and energy release calculations applying the Reference 2 LOCA mass and energy release methodology. (Note: Table 3.1-12, Part 2 lists the 1971 ANS decay heat standard used in the Reference 5 methodology.)

Significant assumptions in the generation of the decay heat curve for use in the LOCA mass and energy release analysis include the following:

- Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

- Decay heat power from the following fissioning isotopes are included: U-238, U-235, and Pu-239.

- Fission rate is constant over the operating history of maximum power level.

- The factor accounting for neutron capture in fission products has been taken from Equation 11 of Reference 6, up to 10,000 seconds and from Table 10 of Reference 6, beyond 10,000 seconds.
- The fuel has been assumed to be at full power for 10^8 seconds.
- The number of atoms of U-239 produced per second has been assumed to be equal to 70 percent of the fission rate.
- The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.
- Two-sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, Safety Evaluation Report (SER) of the March 1979 evaluation model (Reference 2), use of the ANS Standard ANSI/ANS-5.1-1979 decay heat model was approved for the calculation of mass and energy releases to the containment following a LOCA.

3.1.5.5 Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, the EPITOME code continues the FROTH calculation for SG cooldown removing steam generator secondary energy at different rates (i.e., first and second stage rates). The first stage rate is applied until the steam generator reaches T_{sat} at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the containment pressure. Then the second stage rate is used until the final depressurization, when the secondary reaches the reference temperature of T_{sat} at 14.7 psia, or 212°F. The heat removal of the broken-loop and intact-loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature, and a secondary side heat transfer coefficient determined using a modified McAdam's correlation. Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The SG energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. The intermediate equilibrium pressures are chosen as discussed in Reference 2, Sections 2.3 and 3.3. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user-specified time of the final depressurization at 212°F. With current methodology (Reference 2), all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds, i.e., 14.7 psia and 212°F.

3.1.5.6 Long Term Mass & Energy Releases

The long-term (greater than 3600 seconds) mass and energy release calculations are performed through user defined input functions which is an option in the MAAP-DBA code. The MAAP-DBA code was used for convenience. This method of determining the long-term mass and energy releases is consistent with past applications of Westinghouse methodology. See Section 4.1.2 for discussion of long term M&E calculations.

3.1.5.7 Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are given in Tables 3.1-13, and 3.1-14 and 3.1-15 for BVPS-1 and in Tables 3.1-39, and 3.1-40 and 3.1-41 and 3.1-48 for BVPS-2. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Tables 3.1-16, 3.1-17 and 3.1-18 for BVPS-1 and in Tables 3.1-42, and 3.1-43 and 3.1-44 and 3.1-49 for BVPS-2. The energy sources are listed below.

- RCS water
- Accumulator water (all three inject)
- Pumped SI water
- Decay heat
- Core stored energy
- RCS metal (includes the reactor vessel and internals, hot and cold leg piping, SG inlet and outlet plenums, and SG tubes)
- SG metal (includes transition cone, shell, wrapper, and other internals)

  Note: The DEHL cases also conservatively include the upper internals and upper elliptical head.
- SG secondary energy (includes fluid mass and steam mass)
- Secondary transfer of energy (feedwater into, and steam out of, the SG secondary)

The energy reference points are as follows.

- Available energy: 212°F; 14.7 psia
- Total energy content: 32°F; 14.7 psia

The mass and energy inventories are presented at the following times, as appropriate:

- Time zero (initial conditions)
- End of blowdown time
- End of refill time
- End of reflood time
- Time of broken loop steam generator equilibration to pressure setpoint
- Time of intact loop steam generator equilibration to pressure setpoint
- Time of full depressurization (3600 seconds)

The Zirc-water reaction energy was not considered in the mass and energy release data presented because the clad temperature was not assumed to increase high enough for the rate of the Zirc-water reaction to be of any significance.

For BVPS-1, the analyses described in the previous sections were based on a maximum RWST temperature of 65°F, and a maximum sump recirculation water temperature of 190°F for the minimum ECCS case and 210°F for the maximum ECCS case. For BVPS-2, the analyses described in the previous sections were based on a maximum RWST temperature of 65°F, and a maximum recirculation water temperature of 120°F for the minimum ECCS case, 120°F for the CIB failure case, and 150°F for the Service Water (SW) failure case. However, at this time the RWST Technical Specification Temperature of 55°F (BVPS-1) and 50°F (BVPS-2) are not being changed.

3.1.6 Additional LOCA Mass and Energy Input to Support the Recirc Spray Pumps and LHSI Pump NPSH Calculations (BVPS-1)

The sump water temperature is an important parameter for determining the minimum available NPSH for the recirc spray pumps and LHSI pump during the recirculation mode of a LOCA transient. The sump water is cooled by the River Water or Service Water (RW/SW) system.

A maximum sump temperature transient is needed to calculate the minimum available NPSH for the recirculation pumps and to determine the capability of providing adequate cooling for vital equipment during recirculation. Also the long term DEHL LOCA case has the potential to maximize the sump water temperature since all of the safety injection (SI) or recirculation flow must pass through the core to absorb decay heat before it spills out the break and enters the containment sump.

The LOCA mass and energy release response calculations discussed in this section supplement the releases discussed previously. The additional calculations were conservatively performed for a DEHL break with both minimum and maximum safety injection assumptions using the methodology outlined in Reference 5, and apply to both units. This method applies the 1971 ANS decay heat standard. See Table 3.1-12, Part 2.

The DEHL blowdown mass and energy releases are provided in Table 3.1-19. The DEHL reflood and post-reflood mass and energy releases for the DEHL cases with minimum and maximum SI assumptions are provided in Tables 3.1-20 and 21; the mass balances are provided in Tables 3.1-22 and 23; the energy balances are provided in Tables 3.1-24 and 25, and the reflood parameters are provided in Tables 3.1-26 and 27 respectively.

Tables 3.1-19 through Table 3.1-27 present results for Case 1L and Case 2L using the methodology from Reference 5, which results in maximizing the steam releases. These releases were also adjusted to concentrate as much energy as possible in the containment sump by adjusting the releases by:

- adding the decay heat energy and vessel thick metal energy to the sensible heating of the water, and
- maximizing the spillage of hot water to the sump.

These cases are referred to as Case 3L and Case 4L respectively in Table 4-1 and Table 4-23.

The MAAP-DBA Containment Evaluation Model (Chapter 4) was used to calculate the sump temperature. The results of the sump temperature response calculation are presented in Section 4.7.8.

3.1.7 SBLOCA Considerations

A spectrum of IBLOCA and SBLOCA was also evaluated to determine the response of the containment depressurization system and impact on long term core cooling. Specific M&E's for these medium and SBLOCA were derived using the MAAP-DBA computer code and are discussed in detail in Sections 4.7.2 and 9.5.

3.1.8 Conclusions

BVPS-1 and -2 plant specific LOCA mass and energy release analyses were developed using approved design basis methodology. The BVPS-1 analysis bounds core operation at uprated conditions with or without the planned replacement SGs. The BVPS-2 analysis bounds core operation at uprated conditions with the current SGs. The results of this analysis were provided for use in the containment analysis.

The consideration of the various energy sources in the long-term mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied.

3.1.9 References

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

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

3. Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 71062), for D. C. Cook Nuclear Plant Unit-1," June 9, 1989.

4. EPRI 294-2, "Mixing of Emergency Core Cooling Water with Steam; 1/3-Scale Test and Summary," (WCAP-8423), Final Report, June 1975.

5. WCAP-8264-P-A, Rev. 1 (Proprietary), WCAP-8312-A (Nonproprietary), "Westinghouse Mass and Energy Release Data For Containment Design," August 1975.

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

7. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-11873-A, August 1988.

Table 3.1-1
System Parameters Initial Conditions For Thermal Uprate (BVPS-1)

Parameters	Value
Core Thermal Power (MWt)*	2917.4
Reactor Coolant System Total Flowrate (lbm/sec)	27583.3
Vessel Outlet Temperature (°F)*	621.0
Core Inlet Temperature (°F)*	547.1
Vessel Average Temperature (°F)*	584.0
Initial Steam Generator Steam Pressure (psia)	831.0
Steam Generator Design	54F
Steam Generator Tube Plugging (percent)	0
Initial Steam Generator Secondary Side Mass (lbm)*	131011
Assumed Maximum Containment Backpressure (psia)	59.7
Accumulator	
Water Volume (ft^3) per accumulator	1077.6
N_2 Cover Gas Pressure (psia)	575
Temperature (°F)	105

Note:
* The Core Power, RCS Temperature, and Secondary Side Mass values listed above include uncertainty allowance.

Table 3.1-2
Safety Injection Flow Minimum Safeguards (BVPS-1)

RCS Pressure (psig)	Total Flow (GPM)
Injection Mode (Reflood Phase)	
0	3825.5
20	3564.3
50	3131.0
100	2226.5
150	716.2
200	430.0
400	407.4
600	383.7
Cold Leg Recirculation Mode	
0	3072

Notes:

1. A maximum Safety Injection Temperature of 65°F was used during the Injection Phase and 190°F was used during the Recirculation Phase, for Case 6L (Reference 2 Methodology).
2. A conservative maximum safety injection temperature of 105°F was used during the injection phase and 292°F was used during the recirculation phase for Case 1L (Reference 5 Methodology).

Table 3.1-3
Safety Injection Flow Maximum Safeguards (BVPS-1)

RCS Pressure (psig)	Total Flow (GPM)
Injection Mode (Reflood Phase)	
0	5842
50	5092
100	4208
120	3792
200	840
400	807
600	771
Cold Leg Recirculation Mode	
0	5050

Notes:

1. A maximum Safety Injection Temperature of 65°F was used during the Injection Phase and 210°F was conservatively used during the Recirculation Phase, for Case 7L (Reference 2 Methodology).
2. A conservative maximum safety injection temperature of 105°F was used during the injection phase and 292°F was used during the recirculation phase for Case 2L (Reference 5 Methodology).

Table 3.1-4
Double-Ended Hot-Leg Break Blowdown Mass and Energy Releases (BVPS-1) (Case 8L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
.00000	.0	.0	.0	.0
.00109	46610.3	29915.0	46609.0	29913.2
.00217	46171.1	29631.7	45920.8	29466.0
.102	40888.4	26595.7	26652.5	17066.7
.202	34830.1	22703.4	24188.7	15421.5
.301	34365.7	22340.8	21479.3	13551.9
.401	33362.0	21670.6	20111.0	12505.6
.502	32543.3	21137.8	19219.3	11768.5
.602	32400.5	21041.1	18623.3	11240.1
.701	32436.3	21072.0	18078.9	10773.7
.801	32107.7	20896.9	17712.0	10435.1
.902	31571.8	20609.5	17373.3	10132.5
1.00	31218.0	20462.9	17100.0	9886.2
1.10	30928.9	20388.1	16879.7	9682.7
1.20	30649.9	20327.7	16690.5	9508.9
1.30	30289.1	20209.0	16570.8	9382.3
1.40	29888.2	20061.2	16500.6	9291.2
1.50	29438.5	19883.7	16479.3	9232.7
1.60	28922.9	19662.4	16497.3	9200.5
1.70	28364.9	19411.2	16545.5	9188.7
1.80	27764.4	19131.6	16616.8	9192.0
1.90	27156.7	18842.2	16699.0	9204.7
2.00	26532.2	18533.8	16782.8	9221.6
2.10	25871.9	18192.5	16866.1	9241.1
2.20	25212.8	17842.4	16946.6	9262.2
2.30	24578.1	17498.1	17018.4	9281.9
2.40	23935.1	17136.6	17083.1	9300.5
2.50	23325.2	16784.4	17135.3	9315.0

Table 3.1-4 (Continued)
Double-Ended Hot-Leg Break Blowdown Mass and Energy Releases (BVPS-1) (Case 8L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
2.60	22739.8	16439.3	17176.0	9325.6
2.70	22130.0	16060.4	17202.5	9330.7
2.80	21577.5	15708.7	17214.9	9330.2
2.90	21072.8	15379.1	17213.8	9323.9
3.00	20603.5	15058.0	17199.0	9311.7
3.10	20202.3	14772.6	17172.1	9294.2
3.20	19843.0	14503.8	17133.9	9271.6
3.30	19524.4	14252.5	17085.3	9244.3
3.40	19261.1	14031.5	17027.9	9213.0
3.50	19028.5	13822.9	16961.2	9177.4
3.60	18836.3	13637.0	16887.5	9138.5
3.70	18677.3	13468.2	16805.9	9095.8
3.80	18544.7	13313.6	16718.2	9050.2
3.90	18442.3	13177.6	16624.5	9001.7
4.00	18359.6	13052.5	16524.6	8950.1
4.20	18273.5	12852.5	16306.8	8837.9
4.40	18340.4	12746.9	16060.9	8711.3
4.60	18506.2	12704.7	15771.4	8561.9
4.80	18750.4	12712.0	15445.0	8393.7
5.00	19128.9	12776.4	15156.7	8247.6
5.20	11682.8	8813.9	14816.1	8073.4
5.40	14643.7	10645.5	14397.9	7857.2
5.60	14898.3	10688.8	13971.4	7637.6
5.80	15008.0	10665.4	13539.2	7416.2
6.00	15156.1	10667.5	13119.7	7202.8
6.20	15174.3	10585.0	12700.5	6990.0
6.40	15336.2	10598.2	12263.7	6767.8
6.60	15527.5	10585.2	11801.5	6531.1

Table 3.1-4 (Continued) Double-Ended Hot-Leg Break Blowdown Mass and Energy Releases (BVPS-1) (Case 8L)				
Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
6.80	15482.1	10491.5	11349.4	6300.0
7.00	15702.1	10504.6	10916.8	6079.2
7.20	15893.2	10518.8	10485.2	5858.2
7.40	16024.3	10513.6	10064.4	5642.8
7.60	16107.2	10492.1	9668.2	5440.3
7.80	16148.5	10454.6	9293.9	5248.9
8.00	16119.3	10385.1	8944.0	5069.9
8.20	15958.6	10252.6	8609.1	4898.7
8.40	15625.0	10031.1	8294.6	4737.9
8.60	15170.5	9749.6	7995.3	4585.0
8.80	14748.8	9491.8	7712.3	4440.6
9.00	14393.7	9273.6	7436.6	4300.4
9.20	14068.3	9075.3	7179.3	4170.2
9.40	13724.8	8869.0	6930.8	4045.0
9.60	13352.6	8648.4	6691.7	3925.2
9.80	12969.0	8423.4	6464.9	3812.4
10.0	12584.8	8200.0	6245.8	3704.0
10.2	12206.3	7981.6	6032.6	3599.2
10.2	12201.1	7978.7	6029.4	3597.6
10.4	11835.1	7769.2	5827.1	3499.1
10.6	11478.2	7567.2	5632.8	3405.2
10.8	11119.0	7366.3	5441.3	3313.5
11.0	10764.3	7170.4	5257.3	3226.0
11.2	10418.7	6982.3	5081.4	3143.0
11.4	10069.4	6795.1	4908.0	3061.9
11.6	9730.1	6615.6	4742.4	2985.0
11.8	9392.9	6441.0	4581.3	2910.7
12.0	9060.8	6272.5	4427.1	2840.3

Table 3.1-4 (Continued)
Double-Ended Hot-Leg Break Blowdown Mass and Energy Releases (BVPS-1) (Case 8L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
12.2	8730.1	6108.2	4276.9	2771.8
12.4	8377.0	5935.4	4130.8	2705.6
12.6	7976.2	5743.3	3986.3	2640.2
12.8	7525.1	5534.6	3843.8	2575.6
13.0	7008.5	5303.3	3697.5	2509.5
13.2	6434.8	5054.9	3545.5	2441.6
13.4	5813.9	4786.1	3383.2	2370.9
13.6	5180.9	4506.3	3209.9	2297.4
13.8	4555.9	4214.7	3025.3	2221.8
14.0	3937.2	3883.0	2826.6	2142.8
14.2	3302.4	3490.6	2616.7	2061.2
14.4	2792.1	3101.8	2386.7	1972.0
14.6	2428.7	2787.1	2127.2	1870.8
14.8	2143.0	2511.8	1837.1	1775.6
15.0	1902.1	2263.5	1519.1	1678.0
15.2	1702.5	2051.7	1218.4	1482.7
15.4	1507.3	1833.7	996.7	1233.5
15.6	1342.1	1646.4	868.8	1081.0
15.8	1184.2	1464.9	782.2	976.3
16.0	1065.4	1332.2	700.9	876.5
16.2	948.1	1191.8	613.8	768.3
16.4	869.2	1097.1	526.9	660.8
16.6	780.6	987.2	455.6	573.0
16.8	684.1	866.3	395.9	498.7
17.0	596.1	755.8	346.2	436.7
17.2	518.0	657.5	288.9	365.0
17.4	444.3	564.5	238.3	301.8
17.6	371.4	472.2	200.4	254.4

Table 3.1-4 (Continued)
Double-Ended Hot-Leg Break Blowdown Mass and Energy Releases (BVPS-1) (Case 8L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
17.8	297.7	379.0	157.2	200.0
18.0	212.1	270.0	111.5	142.4
18.2	61.3	77.6	55.9	71.8
18.4	278.9	357.9	103.7	133.5
18.6	124.1	159.4	82.5	105.8
18.8	125.0	161.2	103.0	132.4
19.0	177.5	229.6	95.8	123.3
19.2	195.9	253.8	99.0	127.4
19.4	197.0	255.3	102.7	131.9
19.6	178.7	231.8	77.2	99.4
19.8	219.4	283.8	65.5	84.6
20.0	253.2	327.8	73.9	95.8
20.2	321.2	414.7	85.8	111.0
20.4	403.6	517.7	91.4	117.9
20.6	422.6	540.8	110.9	142.8
20.8	450.8	556.9	103.0	132.4
21.0	378.2	473.0	140.9	180.9
21.2	372.1	466.7	95.8	122.9
21.4	470.3	576.9	97.0	124.7
21.6	445.8	554.6	83.6	107.8
21.8	323.4	408.0	78.3	101.0
22.0	91.9	119.4	57.3	74.1
22.2	.0	.0	.0	.0

* Mass and Energy exiting from the RV side of the break
** Mass and Energy exiting from the SG side of the break

Table 3.1-5
Double-Ended Pump Suction Break Blowdown Mass and Energy Releases
(Same for all DEPS Runs) (BVPS-1) (Case 6L & Case 7L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
.00000	.0	.0	.0	.0
.00111	86328.0	46576.4	40275.9	21682.3
.101	40307.9	21771.0	20311.6	10927.4
.202	47126.9	25635.7	22479.2	12101.8
.301	46925.4	25755.9	23012.6	12401.0
.401	47194.3	26187.2	22696.1	12244.5
.501	46805.7	26295.2	21992.2	11874.6
.602	44647.8	25398.6	21311.8	11513.8
.701	44824.0	25789.8	20823.0	11253.7
.801	44733.7	25996.8	20440.3	11050.7
.902	44015.3	25812.1	20143.1	10894.0
1.00	42956.2	25407.3	19924.0	10778.4
1.10	41899.1	24991.3	19757.3	10690.3
1.20	40878.9	24586.9	19662.8	10640.7
1.30	39938.6	24215.4	19641.5	10630.2
1.40	39096.9	23890.2	19657.0	10639.1
1.50	38373.8	23621.3	19640.1	10629.4
1.60	37726.9	23387.2	19589.6	10600.9
1.70	37058.7	23137.4	19540.4	10573.0
1.80	36340.3	22862.4	19516.2	10558.9
1.90	35600.6	22590.9	19471.2	10533.6
2.00	34792.4	22295.7	19360.1	10472.2
2.10	33760.9	21876.9	19182.1	10374.5
2.20	32417.3	21261.7	18960.4	10253.2
2.30	31045.2	20629.4	18605.5	10060.2
2.40	29602.3	19943.2	18277.5	9882.0

Table 3.1-5 (Continued)
Double-Ended Pump Suction Break Blowdown Mass and Energy Releases
(Same for all DEPS Runs) (BVPS-1) (Case 6L & Case 7L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
2.50	27463.0	18739.7	17936.0	9696.5
2.60	23679.2	16340.9	17641.3	9537.0
2.70	21310.6	14895.4	17357.6	9383.7
2.80	19830.4	13995.2	17045.5	9215.4
2.90	18412.1	13061.0	16763.1	9063.7
3.00	17352.9	12355.2	16520.0	8933.8
3.10	16489.3	11779.6	16289.7	8811.0
3.20	15741.3	11280.8	16069.6	8693.8
3.30	15119.1	10871.2	15874.8	8590.9
3.40	14582.9	10522.2	15718.9	8508.6
3.50	14068.0	10184.6	15521.4	8403.6
3.60	13598.1	9878.9	15358.9	8317.9
3.70	13195.3	9620.9	15200.7	8234.4
3.80	12848.5	9398.2	15046.2	8152.9
3.90	12534.6	9192.1	14892.8	8072.0
4.00	12254.3	9004.7	14758.6	8001.5
4.20	11804.1	8699.6	14501.7	7866.7
4.40	11428.7	8440.7	14271.1	7746.3
4.60	11111.7	8222.9	14050.8	7631.4
4.80	10854.3	8047.0	13803.2	7501.7
5.00	10625.4	7891.7	13298.2	7232.2
5.20	10387.8	7734.9	13076.4	7116.6
5.40	10158.5	7588.8	14660.0	7987.7
5.60	9979.2	7472.2	14500.8	7904.9
5.80	9867.5	7399.2	14305.8	7805.4
6.00	10258.8	7702.6	14222.2	7765.8
6.20	10209.2	8045.0	13904.0	7597.7

Table 3.1-5 (Continued) Double-Ended Pump Suction Break Blowdown Mass and Energy Releases (Same for all DEPS Runs) (BVPS-1) (Case 6L & Case 7L)				
	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
6.40	8820.8	7744.0	13698.0	7491.8
6.60	7980.5	7358.6	13464.7	7369.7
6.80	7742.2	7116.4	13146.0	7199.8
7.00	7759.1	6953.3	12832.5	7031.6
7.20	7902.8	6845.9	12642.0	6929.6
7.40	8045.0	6745.7	12550.2	6878.6
7.60	8144.0	6676.1	12413.3	6798.7
7.80	8169.6	6591.9	12200.0	6675.0
8.00	8181.2	6511.3	11982.0	6550.3
8.20	8174.8	6427.7	11826.8	6461.8
8.40	8133.7	6335.0	11671.9	6374.0
8.60	8062.2	6238.2	11478.2	6264.8
8.80	7963.5	6139.8	11293.7	6161.1
9.00	7842.4	6042.5	11137.9	6073.7
9.20	7699.6	5944.5	10969.5	5979.6
9.40	7543.8	5848.6	10792.4	5881.0
9.60	7379.2	5754.7	10630.2	5791.1
9.80	7214.3	5667.4	10466.2	5700.6
10.0	7041.0	5573.6	10287.9	5602.5
10.2	6874.7	5480.9	10118.8	5510.1
10.4	6713.8	5388.1	9952.1	5419.4
10.6	6555.8	5294.2	9781.7	5326.9
10.8	6405.4	5202.4	9617.2	5237.6
11.0	6259.3	5111.1	9455.8	5150.1
11.2	6113.6	5017.1	9295.6	5063.1
11.4	5958.1	4915.0	9106.9	4960.6
11.6	5798.1	4806.1	8935.7	4868.4
11.8	5641.3	4689.2	8769.4	4779.2

Table 3.1-5 (Continued)
Double-Ended Pump Suction Break Blowdown Mass and Energy Releases
(Same for all DEPS Runs) (BVPS-1) (Case 6L & Case 7L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
12.0	5496.8	4571.0	8592.3	4683.6
12.2	5362.4	4454.7	8422.7	4592.4
12.4	5230.1	4338.5	8249.6	4499.4
12.6	5097.1	4222.0	8076.6	4406.5
12.8	4969.7	4110.6	7823.7	4269.0
13.0	4854.9	4009.5	7562.6	4128.6
13.2	4755.8	3921.2	7443.0	4062.0
13.4	4660.4	3841.7	7191.8	3898.8
13.6	4568.5	3773.9	7174.9	3839.4
13.8	4475.3	3714.9	7115.1	3741.6
14.0	4377.6	3662.4	7093.4	3655.2
14.2	4269.4	3614.1	6979.8	3522.7
14.4	4151.4	3570.4	6652.7	3288.2
14.6	4023.0	3533.2	6385.1	3088.0
14.8	3885.7	3499.2	6162.4	2922.3
15.0	3738.5	3469.7	5919.0	2759.5
15.2	3584.2	3445.7	5730.1	2630.4
15.4	3418.0	3424.9	5546.7	2511.2
15.6	3214.1	3381.6	5340.3	2387.9
15.8	2890.7	3246.9	4903.2	2164.0
16.0	2564.9	3058.2	4538.9	1969.6
16.2	2319.6	2840.7	4324.6	1839.9
16.4	2102.1	2594.6	4027.6	1683.4
16.6	1918.7	2377.3	3595.4	1476.0
16.8	1760.6	2187.8	3166.8	1272.4
17.0	1627.6	2027.0	2789.1	1091.8
17.2	1499.8	1871.3	2516.2	958.2
17.4	1379.0	1723.6	2346.9	869.9

Table 3.1-5 (Continued)
Double-Ended Pump Suction Break Blowdown Mass and Energy Releases
(Same for all DEPS Runs) (BVPS-1) (Case 6L & Case 7L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
17.6	1248.8	1563.3	2291.2	828.2
17.8	1136.7	1425.4	2258.2	797.2
18.0	1032.7	1296.6	2304.1	793.3
18.2	825.2	1036.9	2301.4	771.9
18.4	586.6	738.3	1958.1	645.6
18.6	419.1	528.6	1583.8	517.9
18.8	292.7	369.6	1217.3	396.9
19.0	191.8	242.5	972.4	317.2
19.2	117.5	148.8	784.0	256.2
19.4	14.8	18.8	848.2	277.8
19.6	.0	.0	939.8	306.5
19.8	.0	.0	987.4	319.3
20.0	.0	.0	976.1	313.9
20.2	.0	.0	908.2	291.5
20.4	.0	.0	789.9	253.7
20.6	.0	.0	604.1	194.7
20.8	.0	.0	295.9	96.0
21.0	.0	.0	.0	.0

* Mass and Energy exiting the SG side of the break
** Mass and Energy exiting the pump side of the break

Table 3.1-6
Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
21.5	.0	.0	.0	.0
21.7	.0	.0	.0	.0
21.8	.0	.0	.0	.0
21.9	.0	.0	.0	.0
21.9	.0	.0	.0	.0
22.0	70.1	82.6	.0	.0
22.1	30.4	35.8	.0	.0
22.2	32.1	37.8	.0	.0
22.3	37.1	43.7	.0	.0
22.4	46.9	55.2	.0	.0
22.5	54.3	63.9	.0	.0
22.6	61.3	72.2	.0	.0
22.7	67.9	80.0	.0	.0
22.8	72.8	85.8	.0	.0
22.9	77.6	91.4	.0	.0
23.0	81.1	95.5	.0	.0
23.0	82.2	96.8	.0	.0
23.1	86.6	102.0	.0	.0
23.2	90.9	107.0	.0	.0
23.3	94.9	111.8	.0	.0
23.4	98.9	116.5	.0	.0
23.5	102.7	121.0	.0	.0
23.6	106.4	125.3	.0	.0
23.7	110.0	129.5	.0	.0
23.8	113.5	133.7	.0	.0
23.9	116.9	137.7	.0	.0
24.0	120.2	141.6	.0	.0

Table 3.1-6 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
25.0	149.6	176.3	.0	.0
26.1	444.3	526.3	4349.7	497.9
26.7	445.8	528.2	4348.1	504.3
27.1	442.6	524.3	4317.1	502.1
28.1	433.1	512.9	4224.7	494.5
29.1	423.0	500.8	4126.1	485.9
30.1	413.0	488.9	4027.4	477.1
31.2	428.2	507.0	4198.7	476.7
31.3	427.2	505.9	4189.5	475.9
32.2	419.1	496.2	4109.6	468.5
33.2	410.4	485.9	4023.7	460.5
34.2	402.2	476.0	3941.1	452.8
35.2	394.3	466.6	3861.6	445.4
36.2	386.8	457.7	3785.2	438.2
37.2	379.6	449.1	3711.7	431.3
38.2	372.8	441.0	3641.0	424.7
39.2	366.2	433.2	3573.0	418.3
40.2	360.0	425.7	3507.3	412.2
41.2	354.0	418.6	3444.0	406.2
42.2	348.2	411.8	3382.8	400.5
43.2	342.7	405.2	3323.7	394.9
44.1	337.9	399.5	3272.1	390.1
44.2	337.4	398.9	3266.5	389.6
45.2	332.3	392.8	3211.1	384.4
46.2	327.4	387.0	3157.4	379.3
47.2	322.6	381.3	3105.3	374.4
48.2	318.1	375.9	3054.7	369.6
49.2	313.6	370.6	3005.5	364.9

Table 3.1-6 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
50.2	309.3	365.5	2957.7	360.4
51.2	305.2	360.6	2911.2	356.0
51.8	302.8	357.7	2883.8	353.4
52.2	301.2	355.8	2865.8	351.7
53.2	297.3	351.2	2821.7	347.5
54.2	293.5	346.7	2778.6	343.4
55.2	289.8	342.3	2736.5	339.4
56.2	231.3	272.9	1995.3	273.5
57.2	260.8	307.8	237.8	116.6
58.2	261.3	308.5	237.9	116.9
59.2	256.3	302.6	236.1	114.4
60.2	251.4	296.7	234.3	112.0
61.2	246.5	291.0	232.6	109.7
62.2	241.7	285.3	230.9	107.4
63.2	236.8	279.5	229.2	105.1
64.2	232.4	274.2	227.6	103.0
65.2	228.1	269.2	226.1	101.0
66.2	224.0	264.3	224.7	99.1
67.2	220.0	259.6	223.3	97.2
68.2	216.1	255.0	222.0	95.4
69.2	212.4	250.5	220.7	93.7
70.2	208.7	246.2	219.4	92.1
70.4	208.0	245.3	219.1	91.7
71.2	205.1	242.0	218.2	90.4
72.2	201.7	237.9	217.0	88.9
73.2	198.3	233.9	215.8	87.4
74.2	195.0	230.0	214.7	85.9
75.2	191.9	226.3	213.6	84.5

Table 3.1-6 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
76.2	188.8	222.6	212.6	83.2
77.2	185.8	219.1	211.5	81.9
78.2	182.9	215.6	210.6	80.6
79.2	180.0	212.3	209.6	79.4
80.2	177.3	209.0	208.7	78.2
81.2	174.6	205.9	207.8	77.1
82.2	172.1	202.9	207.0	76.0
84.2	167.2	197.1	205.4	73.9
86.2	162.6	191.7	203.9	72.0
88.2	158.4	186.7	202.5	70.3
90.2	154.4	182.0	201.3	68.7
92.2	150.7	177.7	200.1	67.2
94.2	147.4	173.7	199.1	65.8
94.3	147.2	173.5	199.0	65.8
96.2	144.3	170.0	198.1	64.6
98.2	141.4	166.7	197.2	63.5
100.2	138.8	163.6	196.4	62.4
102.2	136.4	160.7	195.7	61.5
104.2	134.2	158.2	195.0	60.6
106.2	132.2	155.8	194.4	59.9
108.2	130.5	153.7	193.9	59.2
110.2	128.9	151.8	193.4	58.5
112.2	127.4	150.1	192.9	58.0
114.2	126.1	148.6	192.5	57.5
116.2	125.0	147.2	192.2	57.0
118.2	124.0	146.0	191.9	56.6
120.2	123.1	145.0	191.6	56.3
122.2	122.3	144.1	191.4	55.9

Table 3.1-6 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
123.3	121.9	143.6	191.2	55.8
124.2	121.6	143.3	191.1	55.7
126.2	121.0	142.6	190.9	55.4
128.2	120.5	142.0	190.8	55.2
130.2	120.1	141.5	190.6	55.0
132.2	119.7	141.1	190.5	54.9
134.2	119.5	140.8	190.4	54.7
136.2	119.2	140.5	190.3	54.6
138.2	119.1	140.3	190.2	54.5
140.2	119.0	140.2	190.2	54.5
142.2	118.9	140.1	190.1	54.4
144.2	118.9	140.0	190.1	54.3
146.2	118.9	140.0	190.1	54.3
148.2	118.9	140.1	190.0	54.3
150.2	119.0	140.2	190.0	54.3
152.2	119.0	140.3	190.0	54.3
154.2	119.1	140.4	190.0	54.3
155.3	119.2	140.5	190.0	54.3
156.2	119.3	140.5	190.0	54.3
158.2	119.4	140.7	190.1	54.3
160.2	119.6	140.9	190.1	54.3
162.2	119.8	141.1	190.1	54.4
164.2	119.9	141.3	190.1	54.4
166.2	120.1	141.6	190.2	54.4
168.2	120.4	141.8	190.2	54.5
170.2	120.6	142.1	190.2	54.5
172.2	121.3	142.9	190.5	54.8
174.2	121.8	143.6	191.1	55.0

Table 3.1-6 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
176.2	122.3	144.1	192.1	55.4
178.2	122.7	144.6	193.4	55.8
180.2	123.1	145.0	195.0	56.2
182.2	123.3	145.3	196.8	56.7
184.2	123.5	145.6	198.9	57.2
186.2	123.7	145.7	201.1	57.7
188.2	123.7	145.7	203.5	58.2
189.0	123.7	145.7	204.5	58.4
190.2	123.6	145.7	206.0	58.7
192.2	123.5	145.5	208.7	59.3
194.2	123.2	145.2	211.4	59.8
196.2	122.8	144.7	214.3	60.4
198.2	122.3	144.1	217.4	60.9
200.2	121.7	143.4	220.5	61.5
202.2	121.0	142.5	223.8	62.1
204.2	120.1	141.5	227.2	62.6
206.2	119.1	140.3	230.6	63.2
208.2	118.0	139.0	234.2	63.8
210.2	116.7	137.5	237.9	64.3
212.2	115.3	135.8	241.7	64.9
214.2	113.7	134.0	245.6	65.5
216.2	112.0	132.0	249.6	66.1
218.2	110.2	129.8	253.6	66.6
220.2	109.0	128.4	256.8	67.0
222.2	108.6	128.0	258.7	67.0
224.2	108.3	127.5	260.5	67.0
225.8	107.9	127.2	261.8	67.0

* Mass and Energy exiting the SG side of the break
** Mass and Energy exiting the pump side of the break

Table 3.1-7
Double-Ended Pump Suction Break - Minimum Safeguards
Principle Parameters During Reflood (BVPS-1) (Case 6L)

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
21.0	221.0	.000	.000	.00	.00	.333	.0	.0	.0	.00
21.7	217.0	22.982	.000	.54	1.98	.000	7125.9	7125.9	.0	74.49
21.9	213.6	28.149	.000	1.07	1.93	.000	7029.8	7029.8	.0	74.49
23.0	211.4	2.916	.313	1.50	6.03	.426	6674.2	6674.2	.0	74.49
23.9	210.9	2.797	.442	1.64	9.78	.455	6432.7	6432.7	.0	74.49
26.1	209.4	5.028	.617	1.92	15.62	.687	5432.3	5432.3	.0	74.49
26.7	208.8	4.811	.647	2.01	15.63	.684	5301.0	5301.0	.0	74.49
27.1	208.5	4.679	.662	2.06	15.63	.684	5230.7	5230.7	.0	74.49
30.1	206.6	4.099	.715	2.39	15.63	.677	4788.1	4788.1	.0	74.49
31.2	206.1	4.154	.724	2.50	15.63	.683	4969.0	4581.7	.0	71.27
31.3	206.0	4.143	.725	2.51	15.63	.683	4957.1	4569.6	.0	71.26
37.2	204.5	3.658	.746	3.01	15.63	.668	4367.9	3967.3	.0	70.71
44.1	204.3	3.309	.754	3.50	15.63	.652	3852.4	3441.3	.0	70.09
51.8	205.2	3.033	.756	4.00	15.63	.635	3405.9	2986.4	.0	69.41
56.2	206.1	2.545	.752	4.26	15.63	.582	2409.9	1975.5	.0	67.05
57.2	206.3	2.756	.755	4.32	15.54	.606	428.9	.0	.0	33.22
61.2	207.6	2.613	.754	4.54	14.89	.602	430.7	.0	.0	33.22

Table 3.1-7 (Continued)
Double-Ended Pump Suction Break - Minimum Safeguards
Principle Parameters During Reflood (BVPS-1) (Case 6L)

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
70.4	212.2	2.282	.750	5.00	13.79	.584	436.2	.0	.0	33.22
82.2	220.1	1.975	.747	5.53	12.99	.561	439.9	.0	.0	33.22
94.3	228.8	1.763	.745	6.00	12.69	.539	442.1	.0	.0	33.22
110.2	238.3	1.604	.746	6.57	12.77	.518	443.6	.0	.0	33.22
123.3	244.8	1.540	.748	7.00	13.06	.509	444.1	.0	.0	33.22
140.2	252.0	1.505	.752	7.54	13.57	.506	444.4	.0	.0	33.22
155.3	257.6	1.497	.757	8.00	14.09	.507	444.4	.0	.0	33.22
162.2	259.9	1.496	.759	8.21	14.33	.508	444.4	.0	.0	33.22
172.2	263.0	1.501	.763	8.51	14.68	.510	444.4	.0	.0	33.22
182.2	265.9	1.508	.766	8.80	15.00	.514	444.2	.0	.0	33.22
189.0	267.8	1.504	.769	9.00	15.18	.515	444.2	.0	.0	33.22
208.2	272.5	1.442	.773	9.54	15.50	.509	444.6	.0	.0	33.22
225.8	276.1	1.348	.776	10.00	15.60	.497	445.2	.0	.0	33.22

Table 3.1-8
Double-Ended Pump Suction Break Minimum
Safeguards Post-Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
225.9	126.9	159.6	318.6	82.5
230.9	126.6	159.2	319.0	82.5
235.9	127.2	160.0	318.3	82.1
240.9	126.9	159.6	318.7	82.0
245.9	126.5	159.1	319.0	82.0
250.9	127.2	159.9	318.4	81.6
255.9	126.8	159.5	318.7	81.6
260.9	126.4	159.0	319.1	81.5
265.9	126.1	158.5	319.5	81.4
270.9	126.7	159.4	318.8	81.1
275.9	126.3	158.9	319.2	81.0
280.9	126.0	158.4	319.6	81.0
285.9	125.6	157.9	319.9	80.9
290.9	126.2	158.7	319.3	80.5
295.9	125.9	158.3	319.7	80.5
300.9	125.5	157.8	320.1	80.4
305.9	126.1	158.6	319.4	80.1
310.9	125.7	158.1	319.8	80.0
315.9	125.3	157.6	320.2	79.9
320.9	124.9	157.1	320.6	79.9
325.9	125.5	157.9	320.0	79.5
330.9	125.2	157.4	320.4	79.5
335.9	124.8	156.9	320.8	79.4
340.9	125.4	157.6	320.2	79.1
345.9	125.0	157.1	320.6	79.0
350.9	124.6	156.6	321.0	78.9
355.9	125.1	157.4	320.4	78.6

Table 3.1-8 (Continued)
Double-Ended Pump Suction Break Minimum
Safeguards Post-Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
360.9	124.7	156.9	320.8	78.5
365.9	124.3	156.3	321.2	78.5
370.9	124.9	157.1	320.6	78.2
375.9	124.5	156.5	321.1	78.1
380.9	124.1	156.0	321.5	78.0
385.9	124.6	156.7	320.9	77.7
390.9	124.2	156.2	321.3	77.6
395.9	123.8	155.7	321.8	77.6
400.9	124.3	156.4	321.2	77.3
405.9	124.0	156.0	321.5	77.2
410.9	123.7	155.6	321.8	77.1
415.9	123.4	155.2	322.1	77.0
420.9	124.1	156.0	321.5	76.6
425.9	123.8	155.6	321.8	76.5
430.9	123.4	155.2	322.1	76.5
435.9	123.1	154.8	322.4	76.4
440.9	123.7	155.6	321.8	76.0
445.9	123.4	155.2	322.1	75.9
450.9	123.1	154.8	322.4	75.8
455.9	123.7	155.6	321.8	78.4
460.9	123.4	155.1	322.2	78.3
465.9	123.0	154.7	322.5	78.2
470.9	122.7	154.3	322.8	78.1
475.9	123.3	155.0	322.2	77.8
480.9	123.0	154.6	322.6	77.7
485.9	122.6	154.2	322.9	77.5
490.9	123.2	154.9	322.4	77.2
495.9	122.8	154.5	322.7	77.1

Table 3.1-8 (Continued)
Double-Ended Pump Suction Break Minimum
Safeguards Post-Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
500.9	122.5	154.0	323.1	77.0
505.9	123.0	154.7	322.5	76.7
510.9	122.6	154.2	322.9	76.6
515.9	122.3	153.8	323.2	76.5
520.9	122.8	154.4	322.7	76.2
525.9	122.4	154.0	323.1	76.1
530.9	122.1	153.5	323.5	76.0
535.9	122.6	154.1	323.0	75.6
540.9	122.2	153.7	323.4	75.6
545.9	122.7	154.2	322.9	75.2
550.9	122.3	153.8	323.3	75.1
555.9	121.9	153.3	323.7	75.0
560.9	122.3	153.8	323.2	74.7
565.9	121.9	153.3	323.6	74.6
570.9	122.4	153.9	323.2	74.3
575.9	121.9	153.4	323.6	74.2
580.9	121.5	152.8	324.0	74.1
585.9	121.9	153.3	323.6	73.8
590.9	121.5	152.8	324.0	73.8
595.9	121.9	153.3	323.6	76.2
600.9	121.5	152.8	324.1	76.1
605.9	121.9	153.3	323.7	75.7
610.9	121.5	152.8	324.1	75.6
615.9	121.8	153.2	323.7	75.3
620.9	121.4	152.7	324.1	75.2
625.9	121.8	153.2	323.7	74.9
630.9	121.4	152.6	324.2	74.8
635.9	121.7	153.0	323.8	74.5

Table 3.1-8 (Continued)
Double-Ended Pump Suction Break Minimum
Safeguards Post-Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
640.9	121.2	152.5	324.3	74.4
645.9	121.6	152.9	324.0	74.1
650.9	121.1	152.3	324.4	74.0
655.9	121.4	152.6	324.2	73.7
660.9	120.9	152.0	324.6	73.6
665.9	121.2	152.4	324.4	73.3
670.9	121.4	152.7	324.1	73.1
675.9	120.9	152.0	324.7	73.0
680.9	121.1	152.3	324.5	72.7
685.9	121.3	152.5	324.3	72.4
690.9	120.7	151.8	324.8	74.9
695.9	120.9	152.0	324.7	74.7
700.9	121.0	152.2	324.5	74.4
705.9	121.1	152.3	324.4	74.1
710.9	120.5	151.6	325.0	74.0
715.9	120.6	151.7	324.9	73.8
720.9	120.6	151.7	324.9	73.5
725.9	120.7	151.7	324.9	73.3
730.9	120.7	151.7	324.9	73.0
735.9	120.6	151.7	324.9	72.8
740.9	120.6	151.6	325.0	72.6
745.9	120.5	151.5	325.1	72.4
750.9	120.3	151.3	325.2	72.1
755.9	120.2	151.1	325.4	71.9
760.9	120.0	150.9	325.5	71.7
765.9	120.4	151.4	325.2	73.9
770.9	120.1	151.0	325.4	73.7
775.9	120.4	151.4	325.2	73.3

Table 3.1-8 (Continued)
Double-Ended Pump Suction Break Minimum
Safeguards Post-Reflood Mass and Energy Releases (BVPS-1) (Case 6L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
780.9	120.0	150.9	325.6	73.2
785.9	120.1	151.1	325.4	72.9
790.9	120.2	151.1	325.4	72.6
795.9	120.2	151.1	325.4	72.3
800.9	120.1	151.0	325.5	72.1
805.9	119.9	150.8	325.6	71.9
810.9	119.6	150.5	325.9	71.7
815.9	119.8	150.6	325.8	71.3
820.9	119.8	150.6	325.8	71.1
825.9	119.6	150.4	326.0	73.3
830.9	119.7	150.5	325.9	72.9
835.9	119.5	150.3	326.0	72.7
840.9	119.5	150.3	326.0	72.4
845.9	119.5	150.3	326.0	72.1
850.9	119.4	150.2	326.1	71.8
855.9	66.2	83.2	379.3	85.5
1273.6	66.2	83.2	379.3	85.5
1273.7	64.6	80.7	381.0	81.5
1275.9	64.5	80.7	381.0	81.2
1547.9	64.5	80.7	381.0	81.2
1548.0	57.9	66.7	387.6	12.9
2900.0	50.4	58.0	395.1	13.1
2900.1	52.1	60.0	355.8	60.9
3600.0	48.6	55.9	359.3	61.5

* Mass and Energy exiting the SG side of the break

** Mass and Energy exiting the pump side of the break

Table 3.1-9
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
21.5	.0	.0	.0	.0
21.7	.0	.0	.0	.0
21.8	.0	.0	.0	.0
21.9	.0	.0	.0	.0
21.9	.0	.0	.0	.0
22.0	70.1	82.6	.0	.0
22.1	30.4	35.8	.0	.0
22.2	32.1	37.8	.0	.0
22.3	37.1	43.7	.0	.0
22.4	46.9	55.2	.0	.0
22.5	54.3	63.9	.0	.0
22.6	61.3	72.2	.0	.0
22.7	67.9	80.0	.0	.0
22.8	72.8	85.8	.0	.0
22.9	77.6	91.4	.0	.0
23.0	81.1	95.5	.0	.0
23.0	82.2	96.8	.0	.0
23.1	86.6	102.0	.0	.0
23.2	90.9	107.0	.0	.0
23.3	94.9	111.8	.0	.0
23.4	98.9	116.5	.0	.0
23.5	102.7	121.0	.0	.0
23.6	106.4	125.3	.0	.0
23.7	110.0	129.5	.0	.0
23.8	113.5	133.7	.0	.0
23.9	116.9	137.7	.0	.0
24.0	120.2	141.6	.0	.0

Table 3.1-9 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
25.0	149.6	176.3	.0	.0
26.1	444.3	526.3	4349.7	497.9
26.7	445.8	528.2	4348.1	504.3
27.1	442.6	524.3	4317.1	502.1
28.1	433.1	512.9	4224.7	494.5
29.1	423.0	500.8	4126.1	485.9
30.1	413.0	488.9	4027.4	477.1
31.2	445.9	528.2	4384.6	482.6
31.3	444.9	527.1	4375.5	481.8
32.2	436.8	517.4	4297.0	474.5
33.2	428.1	507.0	4212.4	466.6
34.2	419.9	497.1	4130.9	459.0
35.2	412.0	487.7	4052.4	451.6
36.2	404.4	478.7	3977.0	444.5
37.0	398.6	471.8	3918.7	439.1
37.2	397.2	470.1	3904.4	437.7
38.2	390.4	461.9	3834.6	431.2
39.2	383.8	454.1	3767.3	424.8
40.2	377.5	446.6	3702.4	418.7
41.2	371.5	439.4	3639.8	412.8
42.2	365.7	432.5	3579.3	407.1
43.2	360.1	425.9	3520.8	401.6
43.7	357.4	422.7	3492.2	399.0
44.2	354.8	419.6	3464.2	396.3
45.2	349.6	413.4	3409.4	391.1
46.2	344.7	407.5	3356.3	386.1
47.2	339.9	401.8	3304.8	381.2
48.2	335.3	396.3	3254.7	376.5

Table 3.1-9 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
49.2	330.8	391.0	3206.1	371.9
50.2	326.4	385.8	3158.8	367.4
51.1	322.6	381.3	3117.2	363.5
51.2	322.2	380.8	3112.8	363.0
52.2	318.2	376.0	3067.9	358.7
53.2	314.2	371.3	3024.2	354.6
54.2	310.4	366.8	2981.6	350.5
55.2	306.7	362.3	2940.0	346.5
56.2	247.7	292.4	2222.8	283.2
57.2	170.1	200.6	402.3	101.8
58.2	158.8	187.1	428.7	96.9
59.2	158.4	186.7	429.4	96.7
60.2	158.1	186.4	430.0	96.5
61.2	157.8	186.0	430.7	96.3
62.2	157.4	185.6	431.4	96.2
63.2	157.1	185.2	432.1	96.0
64.2	156.8	184.8	432.8	95.9
65.2	156.5	184.4	433.5	95.7
66.2	156.2	184.1	434.2	95.5
67.2	155.8	183.7	434.9	95.4
68.2	155.5	183.3	435.6	95.2
69.2	155.2	183.0	436.3	95.1
70.2	154.9	182.6	437.0	94.9
71.2	154.6	182.2	437.7	94.8
71.5	154.5	182.1	437.9	94.7
72.2	154.3	181.8	438.4	94.6
73.2	154.0	181.5	439.1	94.5
74.2	153.7	181.1	439.8	94.3

Table 3.1-9 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
75.2	153.4	180.8	440.5	94.2
76.2	153.0	180.4	441.2	94.0
77.2	152.7	180.0	441.9	93.9
78.2	152.4	179.7	442.6	93.8
79.2	152.1	179.3	443.3	93.6
80.2	151.8	178.9	444.0	93.5
81.2	151.5	178.6	444.7	93.3
82.2	151.2	178.2	445.4	93.2
84.2	150.6	177.5	446.9	92.9
86.2	150.0	176.8	448.3	92.6
88.2	149.4	176.1	449.7	92.4
90.2	148.8	175.3	451.2	92.1
92.2	148.2	174.6	452.7	91.8
94.2	147.5	173.9	454.1	91.6
95.8	147.1	173.3	455.3	91.4
96.2	146.9	173.2	455.6	91.3
98.2	146.3	172.4	457.0	91.1
100.2	145.7	171.7	458.5	90.8
102.2	145.1	171.0	459.9	90.5
104.2	144.5	170.2	461.4	90.3
106.2	143.8	169.5	462.8	90.0
108.2	143.2	168.8	464.2	89.7
110.2	142.6	168.0	465.7	89.5
112.2	141.9	167.3	467.1	89.2
114.2	141.3	166.5	468.5	88.9
116.2	140.7	165.8	469.9	88.7
118.2	140.0	165.0	471.3	88.4
120.2	139.4	164.3	472.7	88.1

Table 3.1-9 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
122.2	138.7	163.5	474.2	87.9
122.3	138.7	163.5	474.2	87.8
124.2	138.1	162.7	475.6	87.6
126.2	137.4	162.0	477.0	87.3
128.2	136.8	161.2	478.4	87.0
130.2	136.1	160.4	479.7	86.8
132.2	135.5	159.7	481.1	86.5
134.2	134.8	158.9	482.5	86.2
136.2	134.2	158.1	483.9	86.0
138.2	133.5	157.4	485.3	85.7
140.2	132.9	156.6	486.7	85.4
142.2	132.2	155.8	488.1	85.2
144.2	131.6	155.0	489.4	84.9
146.2	130.9	154.2	490.8	84.6
148.2	130.2	153.5	492.2	84.4
150.2	129.6	152.7	493.5	84.1
151.6	129.1	152.1	494.5	83.9
152.2	128.9	151.9	494.9	83.8
154.2	128.2	151.1	496.3	83.6
156.2	127.6	150.3	497.6	83.3
158.2	126.9	149.5	499.0	83.1
160.2	126.2	148.7	500.4	82.8
162.2	125.5	147.9	501.7	82.5
164.2	124.8	147.1	503.1	82.3
166.2	124.2	146.3	504.5	82.0
168.2	123.5	145.5	505.8	81.7
170.2	122.8	144.7	507.2	81.5
172.2	122.1	143.9	508.5	81.2

Table 3.1-9 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
174.2	121.4	143.1	509.9	81.0
176.2	120.7	142.3	511.2	80.7
178.2	120.1	141.5	512.6	80.5
180.2	119.4	140.6	513.9	80.2
182.2	118.8	140.0	515.0	80.1
184.2	118.3	139.4	515.9	80.1
184.3	118.3	139.4	515.9	80.1
186.2	117.8	138.8	516.8	80.1
188.2	117.4	138.3	517.7	80.1
190.2	116.9	137.7	518.6	80.0
192.2	116.4	137.2	519.5	80.0
194.2	116.0	136.6	520.3	80.0
196.2	115.5	136.1	521.2	80.0
198.2	115.0	135.5	522.1	79.9
200.2	114.6	135.0	523.0	79.9
202.2	114.1	134.4	523.9	79.9
204.2	113.6	133.9	524.7	79.8
206.2	113.2	133.3	525.6	79.8
208.2	112.7	132.8	526.5	79.8
210.2	112.2	132.2	527.3	79.7
212.2	111.8	131.7	528.2	79.7
214.2	111.3	131.2	529.1	79.6
216.2	110.9	130.6	529.9	79.6
218.2	110.4	130.1	530.8	79.6
220.2	110.0	129.6	531.6	79.5
221.3	109.7	129.3	532.1	79.5

* Mass and Energy exiting the SG side of the break

** Mass and Energy exiting the pump side of the break

Table 3.1-10
Double-Ended Pump Suction Break - Maximum Safeguards Principle Parameters During Reflood (BVPS-1) (Case 7L)

	Flooding						Total	Injection Accum	Spill	
Time (sec)	Temp (°F)	Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	(lbm/sec)			Enthalpy (Btu/lbm)
21.0	221.0	.000	.000	.00	.00	.333	.0	.0	.0	.00
21.7	217.0	22.982	.000	.54	1.98	.000	7125.9	7125.9	.0	74.49
21.9	213.6	28.149	.000	1.07	1.93	.000	7029.8	7029.8	.0	74.49
23.0	211.4	2.916	.313	1.50	6.03	.426	6674.2	6674.2	.0	74.49
23.9	210.9	2.797	.442	1.64	9.78	.455	6432.7	6432.7	.0	74.49
26.1	209.4	5.028	.617	1.92	15.62	.687	5432.3	5432.3	.0	74.49
26.7	208.8	4.811	.647	2.01	15.63	.684	5301.0	5301.0	.0	74.49
27.1	208.5	4.679	.662	2.06	15.63	.684	5230.7	5230.7	.0	74.49
30.1	206.6	4.099	.715	2.39	15.63	.677	4788.1	4788.1	.0	74.49
31.2	206.0	4.283	.724	2.50	15.63	.690	5183.3	4527.6	.0	69.27
31.3	206.0	4.271	.725	2.51	15.63	.690	5171.4	4515.5	.0	69.26
37.0	204.3	3.792	.747	3.01	15.63	.676	4604.2	3935.6	.0	68.50
43.7	203.9	3.443	.754	3.51	15.63	.661	4102.0	3422.8	.0	67.66
51.1	204.6	3.169	.757	4.00	15.63	.647	3669.1	2981.7	.0	69.41
57.2	205.7	2.126	.744	4.38	15.63	.494	714.3	.0	.0	33.22
60.2	206.5	2.069	.744	4.51	15.63	.494	717.3	.0	.0	33.22
71.6	211.6	2.015	.745	5.00	15.63	.495	717.2	.0	.0	33.22

Table 3.1-10 (Continued)
Double-Ended Pump Suction Break - Maximum Safeguards Principle Parameters During Reflood (BVPS-1) (Case 7L)

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Injection Spill (lbm/sec)	Enthalpy (Btu/lbm)
84.2	219.6	1.954	.748	5.53	15.63	.496	717.2	.0	.0	33.22
95.8	227.9	1.897	.750	6.00	15.63	.497	717.2	.0	.0	33.22
110.2	237.3	1.828	.754	6.56	15.63	.498	717.2	.0	.0	33.22
122.3	243.9	1.770	.756	7.00	15.63	.499	717.2	.0	.0	33.22
138.2	251.4	1.695	.759	7.56	15.63	.499	717.3	.0	.0	33.22
151.6	256.8	1.633	.762	8.00	15.63	.499	717.3	.0	.0	33.22
168.2	262.5	1.556	.765	8.52	15.63	.499	717.3	.0	.0	33.22
184.3	267.2	1.485	.767	9.00	15.63	.500	717.4	.0	.0	33.22
202.2	271.6	1.420	.771	9.50	15.63	.502	717.3	.0	.0	33.22
221.3	275.6	1.352	.775	10.00	15.63	.504	717.3	.0	.0	33.22

Table 3.1-11
Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(Thousand lbm/sec)	(Thousand Btu/sec)	(Thousand lbm/sec)	(Thousand Btu/sec)
221.4	129.6	163.0	588.5	92.6
226.4	130.3	163.9	587.8	92.2
231.4	129.9	163.4	588.2	92.2
236.4	129.5	162.9	588.6	92.1
241.4	130.2	163.8	587.9	91.8
246.4	129.8	163.3	588.3	91.7
251.4	129.4	162.8	588.7	91.6
256.4	129.0	162.3	589.1	91.5
261.4	129.7	163.2	588.4	91.2
266.4	129.3	162.7	588.8	91.1
271.4	128.9	162.2	589.2	91.0
276.4	129.6	163.0	588.5	90.7
281.4	129.2	162.5	588.9	90.6
286.4	128.8	162.0	589.3	90.6
291.4	128.4	161.5	589.7	90.5
296.4	129.0	162.3	589.1	90.1
301.4	128.6	161.8	589.5	90.1
306.4	128.2	161.3	589.9	90.0
311.4	128.8	162.1	589.3	89.7
316.4	128.4	161.6	589.7	89.6
321.4	128.0	161.1	590.1	89.5
326.4	128.6	161.8	589.5	89.2
331.4	128.2	161.3	589.9	89.1
336.4	127.8	160.8	590.3	89.0
341.4	128.4	161.5	589.7	88.7
346.4	128.0	161.0	590.1	88.6
351.4	127.5	160.5	590.6	88.6

Table 3.1-11 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
356.4	128.1	161.2	590.0	88.2
361.4	127.7	160.6	590.4	88.2
366.4	127.3	160.1	590.8	88.1
371.4	127.8	160.8	590.3	87.8
376.4	127.4	160.2	590.7	87.7
381.4	126.9	159.7	591.2	87.6
386.4	127.5	160.4	590.6	87.3
391.4	127.0	159.8	591.1	87.2
396.4	126.6	159.3	591.5	87.2
401.4	127.1	159.9	591.0	86.8
406.4	126.8	159.5	591.3	86.7
411.4	126.5	159.1	591.6	86.6
416.4	127.1	159.9	591.0	86.3
421.4	126.8	159.5	591.3	86.2
426.4	126.4	159.1	591.7	86.1
431.4	126.1	158.7	592.0	86.0
436.4	126.7	159.4	591.4	85.7
441.4	126.4	159.0	591.7	85.6
446.4	126.0	158.5	592.1	85.5
451.4	126.6	159.3	591.5	85.2
456.4	126.2	158.8	591.9	85.1
461.4	125.9	158.4	592.2	85.0
466.4	126.4	159.1	591.7	84.6
471.4	126.1	158.6	592.0	84.6
476.4	125.7	158.2	592.4	84.5
481.4	126.3	158.8	591.8	84.1
486.4	125.9	158.4	592.2	86.8
491.4	125.5	157.9	592.6	86.7

Table 3.1-11 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
496.4	126.0	158.5	592.1	86.4
501.4	125.6	158.1	592.5	86.3
506.4	125.3	157.6	592.8	86.2
511.4	125.7	158.2	592.4	85.9
516.4	125.4	157.7	592.7	85.8
521.4	125.8	158.3	592.3	85.4
526.4	125.4	157.8	592.7	85.4
531.4	125.0	157.3	593.1	85.3
536.4	125.5	157.8	592.6	84.9
541.4	125.0	157.3	593.1	84.8
546.4	125.5	157.8	592.6	84.5
551.4	125.0	157.3	593.1	84.4
556.4	125.4	157.8	592.7	84.1
561.4	125.0	157.2	593.1	84.0
566.4	125.4	157.7	592.7	83.7
571.4	124.9	157.2	593.2	83.6
576.4	125.3	157.6	592.8	83.3
581.4	124.8	157.0	593.3	83.2
586.4	125.1	157.4	593.0	82.9
591.4	124.7	156.8	593.4	82.9
596.4	125.0	157.2	593.1	82.6
601.4	124.5	156.6	593.6	82.5
606.4	124.8	157.0	593.3	84.8
611.4	124.4	156.4	593.7	84.7
616.4	124.7	156.8	593.4	84.4
621.4	124.2	156.2	593.9	84.3
626.4	124.5	156.6	593.6	84.0
631.4	124.7	156.9	593.4	83.7

Table 3.1-11 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	**(lbm/sec)**	**(Thousand Btu/sec)**	**(lbm/sec)**	**(Thousand Btu/sec)**
636.4	124.2	156.3	593.9	83.6
641.4	124.4	156.5	593.7	83.4
646.4	123.9	155.9	594.2	83.3
651.4	124.1	156.1	594.0	83.0
656.4	124.3	156.3	593.8	82.7
661.4	124.4	156.5	593.7	82.4
666.4	123.8	155.8	594.3	82.4
671.4	124.0	155.9	594.1	82.1
676.4	124.1	156.1	594.0	81.8
681.4	124.1	156.2	594.0	81.6
686.4	124.2	156.2	593.9	81.3
691.4	123.5	155.4	594.6	83.8
696.4	123.5	155.4	594.6	83.6
701.4	123.5	155.4	594.6	83.3
706.4	123.4	155.3	594.7	83.1
711.4	123.4	155.2	594.7	82.8
716.4	123.9	155.9	594.2	82.5
721.4	123.7	155.7	594.4	82.2
726.4	123.5	155.4	594.6	82.0
731.4	123.3	155.1	594.8	81.8
736.4	123.6	155.5	594.5	81.5
741.4	123.3	155.1	594.8	81.3
746.4	123.5	155.4	594.6	81.0
751.4	123.1	154.9	595.0	80.9
756.4	123.2	155.0	594.9	83.0
761.4	123.2	155.0	594.9	82.7
766.4	123.2	155.0	594.9	82.5
771.4	123.0	154.8	595.1	82.2

Table 3.1-11 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases (BVPS-1) (Case 7L)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
776.4	123.3	155.1	594.8	81.9
781.4	122.9	154.6	595.2	81.7
786.4	122.9	154.7	595.2	81.4
791.4	122.8	154.5	595.3	81.2
796.4	122.9	154.7	595.2	80.8
801.4	122.9	154.6	595.2	80.6
806.4	122.5	154.2	595.6	80.4
811.4	122.8	154.4	595.3	80.0
816.4	122.5	154.1	595.6	82.2
821.4	122.5	154.1	595.6	81.9
826.4	66.7	83.9	651.4	96.2
1242.0	66.7	83.9	651.4	96.2
1242.1	64.3	80.5	653.8	92.1
1246.4	64.2	80.4	653.9	91.4
1543.3	64.2	80.4	653.9	91.4
1543.4	56.5	65.0	661.6	22.0
2500.0	50.7	58.3	667.4	22.2
2500.1	53.5	61.5	612.6	113.9
3600.0	47.9	55.1	618.2	114.9

* Mass and Energy exiting the SG side of the break
** Mass and Energy exiting the pump side of the break

Table 3.1-12 (Part 1)
LOCA Mass and Energy Release Analysis ANS 1979 Core Decay Heat Power Fraction

Time (sec)	ANS 1979 Decay Heat Fraction
10	0.053876
15	0.050401
20	0.048018
40	0.042401
60	0.039244
80	0.037065
100	0.035466
150	0.032724
200	0.030936
400	0.027078
600	0.024931
800	0.023389
1000	0.022156
1500	0.019921
2000	0.018315
4000	0.014781
6000	0.013040
8000	0.012000
10000	0.011262
15000	0.010097
20000	0.009350
40000	0.007778
60000	0.006958
80000	0.006424
100000	0.006021
150000	0.005323
400000	0.003770
600000	0.003201
800000	0.002834
1000000	0.002580
2592000	0.001745

Table 3.1-12 (Part 2)
LOCA Mass and Energy Release Analysis ANS 1971 Core Decay Heat Power Fraction

Time (sec)	ANS 1971 Decay Heat Fraction
100	0.043620
200	0.036610
400	0.031470
800	0.025970
1000	0.024220
2000	0.019000
4000	0.015240
8000	0.012150
10000	0.011250
20000	0.009199
40000	0.007579
80000	0.006272
100000	0.005846
200000	0.004659
400000	0.003641
800000	0.002747
1000000	0.002481
2000000	0.001827
4000000	0.001830
8000000	0.0007659
10000000	0.0006659

Table 3.1-13
Double-Ended Hot-Leg Break Mass Balance (BVPS-1) (Case 8L)

		Time (Sec)		
		.00	22.20	22.20*
		Mass (Thousand lbm)		
Initial	In RCS and ACC	621.13	621.13	621.13
Added Mass	Pumped Injection	.00	.00	.00
	Total Added	.00	.00	.00
Total Available		621.13	621.13	621.13
Distribution	Reactor Coolant	420.62	80.27	80.27
	Accumulator	200.51	138.05	138.05
	Total Contents	621.13	218.32	218.32
Effluent	Break Flow	.00	402.79	402.79
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	402.79	402.79
Total Accountable		621.13	621.12	621.12

* This time is the bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

Table 3.1-14
Double-Ended Pump Suction Break Mass Balance Minimum Safeguards
(BVPS-1) (Case 6L)

		Time (Sec)						
		.00	21.00 (1)	21.00 (2)	225.80 (3)	1273.73 (4)	1547.89 (5)	3600.00 (6)
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	621.20	621.20	621.20	621.20	621.20	621.20	621.20
Added Mass	Pumped Injection	.00	.00	.00	85.65	552.50	674.64	1562.59
	Total Added	.00	.00	.00	85.65	552.50	674.64	1562.59
Total Available		621.20	621.20	621.20	706.84	1173.69	1295.84	2183.78
Distribution	Reactor Coolant	420.68	42.50	56.19	101.84	101.84	101.84	101.84
	Accumulator	200.51	159.91	146.22	.00	.00	.00	.00
	Total Contents	621.20	202.42	202.42	101.84	101.84	101.84	101.84
Effluent	Break Flow	.00	418.77	418.77	596.32	1063.17	1185.31	2073.26
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	418.77	418.77	596.32	1063.17	1185.31	2073.26
Total Accountable		621.20	621.18	621.18	698.16	1165.01	1287.15	2175.10

Notes:

(1) End of Blowdown

(2) Bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure

(6) Time at which both SGs equilibrate to 14.7 psia

Table 3.1-15
Double-Ended Pump Suction Break Mass Balance Maximum Safeguards
(BVPS-1) (Case 7L)

		Time (Sec)						
		.00	21.00 (1)	21.00 (2)	221.30 (3)	1242.08 (4)	1543.28 (5)	3600.00 (6)
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	621.20	621.20	621.20	621.20	621.20	621.20	621.20
Added Mass	Pumped Injection	.00	.00	.00	135.94	868.89	1085.19	2504.92
	Total Added	.00	.00	.00	135.94	868.89	1085.19	2504.92
Total Available		621.20	621.20	621.20	757.14	1490.09	1706.38	3126.11
Distribution	Reactor Coolant	420.68	42.50	56.19	102.02	102.02	102.02	102.02
	Accumulator	200.51	159.91	146.22	.00	.00	.00	.00
	Total Contents	621.20	202.42	202.42	102.02	102.02	102.02	102.02
Effluent	Break Flow	.00	418.77	418.77	646.44	1379.39	1595.68	3015.41
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	418.77	418.77	646.44	1379.39	1595.68	3015.41
Total Accountable		621.20	621.18	621.18	748.46	1481.40	1697.70	3117.43

Notes:
(1) End of Blowdown
(2) Bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill
(3) End of Reflood
(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure.
(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure.
(6) Time at which both SGs equilibrate to 14.7 psia.

Table 3.1-16
Double-Ended Hot-Leg Break Energy Balance (BVPS-1) (Case 8L)

		Time (Sec)		
		.00	22.20	22.20*
		Energy (Million Btu)		
Initial Energy	In RCS, Acc, SG	667.41	667.41	667.41
Added Energy	Pumped Injection	.00	.00	.00
	Decay Heat	.00	6.28	6.28
	Heat From Secondary	.00	-.20	-.20
	Total Added	.00	6.08	6.08
Total Available		667.41	673.49	673.49
Distribution	Reactor Coolant	247.64	17.46	17.46
	Accumulator	14.95	10.29	10.29
	Core Stored	22.89	9.64	9.64
	Primary Metal	124.37	116.25	116.25
	Secondary Metal	74.99	74.79	74.79
	Steam Generator	182.56	181.95	181.95
	Total Contents	667.41	410.38	410.38
Effluent	Break Flow	.00	262.62	262.62
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	262.62	262.62
Total Accountable		667.41	673.00	673.00

* This time is the bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

Table 3.1-17
Double-Ended Pump Suction Break Energy Balance Minimum Safeguards
(BVPS-1) (Case 6L)

		Time (Sec)						
		.00	21.00(1)	21.00(2)	225.80(3)	1273.73(4)	1547.89(5)	3600.00(6)
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	663.25	663.25	663.25	663.25	663.25	663.25	663.25
Added Energy	Pumped Injection	.00	.00	.00	2.85	18.35	22.41	87.58
	Decay Heat	.00	5.72	5.72	25.78	96.42	111.84	209.75
	Heat From Secondary	.00	-.84	-.84	-.84	2.96	2.98	2.98
	Total Added	.00	4.88	4.88	27.79	117.74	137.22	300.31
Total Available		663.25	668.13	668.13	691.04	780.99	800.47	963.56
Distribution	Reactor Coolant	247.55	10.59	11.61	26.78	26.78	26.78	26.78
	Accumulator	14.95	11.92	10.90	.00	.00	.00	.00
	Core Stored	22.82	12.24	12.24	3.91	3.20	3.14	2.71
	Primary Metal	120.95	113.82	113.82	93.30	50.68	45.55	39.50
	Secondary Metal	45.38	45.75	45.75	41.87	23.55	20.23	17.60
	Steam Generator	211.59	213.69	213.69	192.31	105.75	90.93	79.36
	Total Contents	663.25	408.01	408.01	358.18	209.96	186.63	165.95
Effluent	Break Flow	.00	259.63	259.63	324.54	562.71	584.39	769.66
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	259.63	259.63	324.54	562.71	584.39	769.66
Total Accountable		663.25	667.65	667.65	682.72	772.67	771.02	935.61

Notes:

(1) End of Blowdown

(2) Bottom of core recovery time. This time is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure

(6) Time at which both SGs equilibrate to 14.7 psia

Table 3.1-18
Double-Ended Pump Suction Break Energy Balance - Maximum Safeguards (BVPS-1) (Case 7L)

		Time (sec)						
		.00	21.00(1)	21.00(2)	221.30(3)	1242.08(4)	1543.28(5)	3600.00(6)
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	663.25	663.25	663.25	663.25	663.25	663.25	663.25
Added Energy	Pumped Injection	.00	.00	.00	4.52	28.86	36.05	189.48
	Decay Heat	.00	5.72	5.72	25.40	94.58	111.58	209.73
	Heat From Secondary	.00	-.84	-.84	-.84	2.87	2.90	2.90
	Total Added	.00	4.88	4.88	29.08	126.31	150.52	402.10
Total Available		663.25	668.13	668.13	692.33	789.56	813.77	1065.36
Distribution	Reactor Coolant	247.55	10.59	11.61	26.80	26.80	26.80	26.80
	Accumulator	14.95	11.92	10.90	.00	.00	.00	.00
	Core Stored	22.82	12.24	12.24	3.91	3.13	3.08	2.71
	Primary Metal	120.95	113.82	113.82	92.86	50.18	44.57	39.47
	Secondary Metal	45.38	45.75	45.75	41.76	23.44	19.78	17.58
	Steam Generator	211.59	213.69	213.69	191.70	105.22	88.86	79.18
	Total Contents	663.25	408.01	408.01	357.04	208.78	183.09	165.73
Effluent	Break Flow	.00	259.63	259.63	326.97	572.46	600.87	870.97
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	259.63	259.63	326.97	572.46	600.87	870.97
Total Accountable		663.25	667.65	667.65	684.01	781.25	783.96	1036.70

Notes:
(1) End of Blowdown.
(2) Bottom of core recovery time. This time is identical to the end of blowdown time due to the assumption of instantaneous refill.
(3) End of Reflood.
(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure.
(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure.
(6) Time at which both SGs equilibrate to 14.7 psia.

Table 3.1-19
Double-Ended Hot Leg Break Blowdown Mass and Energy Releases
(BVPS-1 Case 1L & Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
.00000	.0	.0	.0	.0
.00109	46610.3	29915.0	46609.0	29913.2
.00217	46171.1	29631.7	45920.8	29466.0
.102	40888.4	26595.7	26652.5	17066.7
.202	34830.1	22703.4	24188.7	15421.5
.301	34365.7	22340.8	21479.3	13551.9
.401	33362.0	21670.6	20111.0	12505.6
.502	32543.3	21137.8	19219.3	11768.5
.602	32400.5	21041.1	18623.3	11240.1
.701	32436.3	21072.0	18078.9	10773.7
.801	32107.7	20896.9	17712.0	10435.1
.902	31571.8	20609.5	17373.3	10132.5
1.00	31218.0	20462.9	17100.0	9886.2
1.10	30928.9	20388.1	16879.7	9682.7
1.20	30649.9	20327.7	16690.5	9508.9
1.30	30289.1	20209.0	16570.8	9382.3
1.40	29888.2	20061.2	16500.6	9291.2
1.50	29438.5	19883.7	16479.3	9232.7
1.60	28922.9	19662.4	16497.3	9200.5
1.70	28364.9	19411.2	16545.5	9188.7
1.80	27764.4	19131.6	16616.8	9192.0
1.90	27156.7	18842.2	16699.0	9204.7
2.00	26532.2	18533.8	16782.8	9221.6
2.10	25871.9	18192.5	16866.1	9241.1
2.20	25212.8	17842.4	16946.6	9262.2
2.30	24578.1	17498.1	17018.4	9281.9
2.40	23935.1	17136.6	17083.1	9300.5

Table 3.1-19 (Continued)
Double-Ended Hot Leg Break Blowdown Mass and Energy Releases
(BVPS-1 Case 1L & Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
2.50	23325.2	16784.4	17135.3	9315.0
2.60	22739.8	16439.3	17176.0	9325.6
2.70	22130.0	16060.4	17202.5	9330.7
2.80	21577.5	15708.7	17214.9	9330.2
2.90	21072.8	15379.1	17213.8	9323.9
3.00	20603.5	15058.0	17199.0	9311.7
3.10	20202.3	14772.6	17172.1	9294.2
3.20	19843.0	14503.8	17133.9	9271.6
3.30	19524.4	14252.5	17085.3	9244.3
3.40	19261.1	14031.5	17027.9	9213.0
3.50	19028.5	13822.9	16961.2	9177.4
3.60	18836.3	13637.0	16887.5	9138.5
3.70	18677.3	13468.2	16805.9	9095.8
3.80	18544.7	13313.6	16718.2	9050.2
3.90	18442.3	13177.6	16624.5	9001.7
4.00	18359.6	13052.5	16524.6	8950.1
4.20	18273.5	12852.5	16306.8	8837.9
4.40	18340.4	12746.9	16060.9	8711.3
4.60	18506.2	12704.7	15771.4	8561.9
4.80	18750.4	12712.0	15445.0	8393.7
5.00	19128.9	12776.4	15156.7	8247.6
5.20	11682.8	8813.9	14816.1	8073.4
5.40	14643.7	10645.5	14397.9	7857.2
5.60	14898.3	10688.8	13971.4	7637.6
5.80	15008.0	10665.4	13539.2	7416.2
6.00	15156.1	10667.5	13119.7	7202.8
6.20	15174.3	10585.0	12700.5	6990.0

Table 3.1-19 (Continued)
Double-Ended Hot Leg Break Blowdown Mass and Energy Releases
(BVPS-1 Case 1L & Case 2L)

	Break Path No. 1 Flow*		**Break Path No. 2 Flow****	
Time (sec)	**(lbm/sec)**	**(Thousand Btu/sec)**	**(lbm/sec)**	**(Thousand Btu/sec)**
6.40	15336.2	10598.2	12263.7	6767.8
6.60	15527.5	10585.2	11801.5	6531.1
6.80	15482.1	10491.5	11349.4	6300.0
7.00	15702.1	10504.6	10916.8	6079.2
7.20	15893.2	10518.8	10485.2	5858.2
7.40	16024.3	10513.6	10064.4	5642.8
7.60	16107.2	10492.1	9668.2	5440.3
7.80	16148.5	10454.6	9293.9	5248.9
8.00	16119.3	10385.1	8944.0	5069.9
8.20	15958.6	10252.6	8609.1	4898.7
8.40	15625.0	10031.1	8294.6	4737.9
8.60	15170.5	9749.6	7995.3	4585.0
8.80	14748.8	9491.8	7712.3	4440.6
9.00	14393.7	9273.6	7436.6	4300.4
9.20	14068.3	9075.3	7179.3	4170.2
9.40	13724.8	8869.0	6930.8	4045.0
9.60	13352.6	8648.4	6691.7	3925.2
9.80	12969.0	8423.4	6464.9	3812.4
10.0	12584.8	8200.0	6245.8	3704.0
10.202	12206.3	7981.6	6032.6	3599.2
10.204	12201.1	7978.7	6029.4	3597.6
10.4	11835.1	7769.2	5827.1	3499.1
10.6	11478.2	7567.2	5632.8	3405.2
10.8	11119.0	7366.3	5441.3	3313.5
11.0	10764.3	7170.4	5257.3	3226.0
11.2	10418.7	6982.3	5081.4	3143.0
11.4	10069.4	6795.1	4908.0	3061.9

Table 3.1-19 (Continued)
Double-Ended Hot Leg Break Blowdown Mass and Energy Releases
(BVPS-1 Case 1L & Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
11.6	9730.1	6615.6	4742.4	2985.0
11.8	9392.9	6441.0	4581.3	2910.7
12.0	9060.8	6272.5	4427.1	2840.3
12.2	8730.1	6108.2	4276.9	2771.8
12.4	8377.0	5935.4	4130.8	2705.6
12.6	7976.2	5743.3	3986.3	2640.2
12.8	7525.1	5534.6	3843.8	2575.6
13.0	7008.5	5303.3	3697.5	2509.5
13.2	6434.8	5054.9	3545.5	2441.6
13.4	5813.9	4786.1	3383.2	2370.9
13.6	5180.9	4506.3	3209.9	2297.4
13.8	4555.9	4214.7	3025.3	2221.8
14.0	3937.2	3883.0	2826.6	2142.8
14.2	3302.4	3490.6	2616.7	2061.2
14.4	2792.1	3101.8	2386.7	1972.0
14.6	2428.7	2787.1	2127.2	1870.8
14.8	2143.0	2511.8	1837.1	1775.6
15.0	1902.1	2263.5	1519.1	1678.0
15.2	1702.5	2051.7	1218.4	1482.7
15.4	1507.3	1833.7	996.7	1233.5
15.6	1342.1	1646.4	868.8	1081.0
15.8	1184.2	1464.9	782.2	976.3
16.0	1065.4	1332.2	700.9	876.5
16.2	948.1	1191.8	613.8	768.3
16.4	869.2	1097.1	526.9	660.8
16.6	780.6	987.2	455.6	573.0
16.8	684.1	866.3	395.9	498.7

Table 3.1-19 (Continued)
Double-Ended Hot Leg Break Blowdown Mass and Energy Releases
(BVPS-1 Case 1L & Case 2L)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
17.0	596.1	755.8	346.2	436.7
17.2	518.0	657.5	288.9	365.0
17.4	444.3	564.5	238.3	301.8
17.6	371.4	472.2	200.4	254.4
17.8	297.7	379.0	157.2	200.0
18.0	212.1	270.0	111.5	142.4
18.2	61.3	77.6	55.9	71.8
18.4	278.9	357.9	103.7	133.5
18.6	124.1	159.4	82.5	105.8
18.8	125.0	161.2	103.0	132.4
19.0	177.5	229.6	95.8	123.3
19.2	195.9	253.8	99.0	127.4
19.4	197.0	255.3	102.7	131.9
19.6	178.7	231.8	77.2	99.4
19.8	219.4	283.8	65.5	84.6
20.0	253.2	327.8	73.9	95.8
20.2	321.2	414.7	85.8	111.0
20.4	403.6	517.7	91.4	117.9
20.6	422.6	540.8	110.9	142.8
20.8	450.8	556.9	103.0	132.4
21.0	378.2	473.0	140.9	180.9
21.2	372.1	466.7	95.8	122.9
21.4	470.3	576.9	97.0	124.7
21.6	445.8	554.6	83.6	107.8
21.8	323.4	408.0	78.3	101.0
22.0	91.9	119.4	57.3	74.1
22.2	.0	.0	.0	.0

* Mass and Energy exiting from the RV side of the break
** Mass and Energy exiting from the SG side of the break

Table 3.1-20
Double-Ended Hot Leg Minimum Safeguards Reflood and Post-Reflood Mass and Energy Releases (BVPS-1 Case 1L)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
22.2	.0	.0	.0	.0
22.4	.0	.0	.0	.0
22.5	.0	.0	.0	.0
22.7	693.5	216.5	.0	.0
22.8	476.4	276.4	.0	.0
28.2	1731.7	572.6	3389.0	252.1
32.3	1678.5	568.7	2912.5	216.6
50.0	1456.9	519.7	1646.5	122.3
51.0	1442.9	516.8	1567.1	116.4
56.2	1380.3	502.8	1342.7	99.7
57.7	1361.0	496.7	.0	.0
61.3	1056.0	441.9	.0	.0
68.2	629.1	366.4	.0	.0
89.7	310.8	307.2	.0	.0
100.0	303.8	303.0	.0	.0
157.2	287.4	285.2	.0	.0
157.2	122.2	145.9	.0	.0
200.0	114.0	136.1	325.7	23.8
500.0	82.5	98.5	357.3	26.1
1000.0	64.4	76.8	375.4	27.4
1499.99	57.0	68.0	382.8	27.9
1500.0	68.6	81.9	371.2	97.2

* Mass and Energy exiting from the RV side of the break
** Mass and Energy exiting from the SG side of the break

Table 3.1-21
Double-Ended Hot Leg Maximum Safeguards Reflood and Post-Reflood Mass and Energy Releases
(BVPS-1 Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
22.2	.0	.0	.0	.0
22.4	.0	.0	.0	.0
22.5	15.4	2.9	.0	.0
22.7	755.5	233.7	.0	.0
22.8	508.8	308.1	.0	.0
25.0	1260.4	492.5	.0	.0
27.2	1732.1	587.7	.0	.0
32.1	1678.8	588.0	3203.2	238.0
50.0	1452.8	540.0	1900.3	141.0
50.8	1441.3	537.7	1833.9	136.1
56.2	1376.8	523.9	1610.8	119.5
57.7	1359.4	518.5	.0	.0
67.3	810.8	424.0	.0	.0
74.7	627.5	391.9	.0	.0
100.0	559.4	372.7	.0	.0
129.5	485.6	352.3	.0	.0
129.6	128.9	153.7	.0	.0
200.0	114.0	136.0	592.1	43.2
500.0	82.5	98.4	623.6	45.5
1000.0	64.4	76.7	641.8	46.8
1499.99	57.0	67.9	649.2	47.4
1500.0	68.6	81.8	637.5	166.9

* Mass and Energy exiting from the RV side of the break
** Mass and Energy exiting from the SG side of the break

Table 3.1-22 Double-Ended Hot Leg Break Mass Balance Minimum Safeguards (BVPS-1 Case 1L)					
		Time (Sec)			
		.00	22.20 (1)	157.19 (2)	1500.00 (3)
		Mass (Thousand lbm)			
Initial	In RCS & Accumulator	621.13	621.13	621.13	621.13
Added Mass	Pumped Injection	.00	.00	59.28	649.83
	Total Added	.00	.00	59.28	649.83
Total Available		621.13	621.13	680.41	1270.96
Distribution	Reactor Coolant	420.62	80.27	122.36	130.88
	Accumulator	200.51	138.05	.00	.00
	Total Contents	621.13	218.32	122.36	130.88
Effluent	Break Flow	.00	402.79	494.39	594.11
	ECCS Spill	.00	.00	63.64	545.94
	Total Effluent	.00	402.79	558.03	1140.05
Total Accountable		621.13	621.12	680.39	1270.94

(1) End of Blowdown
(2) End of Reflood
(3) End of Injection Phase

Table 3.1-23
Double-Ended Hot Leg Break Mass Balance Maximum Safeguards (BVPS-1 Case 2L)

		Time (Sec)			
		.00	22.20(1)	129.55(2)	1500.00(3)
		Mass (Thousand lbm)			
Initial	In RCS & Accumulator	621.13	621.13	621.13	621.13
Added Mass	Pumped Injection	.00	.00	75.74	1043.47
	Total Added	.00	.00	75.74	1043.47
Total Available		621.13	621.13	696.87	1664.60
Distribution	Reactor Coolant	420.62	80.27	122.36	130.36
	Accumulator	200.51	138.05	.00	.00
	Total Contents	621.13	218.32	122.92	130.36
Effluent	Break Flow	.00	402.79	501.71	604.90
	ECCS Spill	.00	.00	72.23	929.31
	Total Effluent	.00	402.79	573.94	1534.21
Total Accountable		621.13	621.12	696.85	1664.58

(1) End of Blowdown
(2) End of Reflood
(3) End of Injection Phase

Table 3.1-24
Double-Ended Hot Leg Break Energy Balance Minimum Safeguards (BVPS-1 Case 1L)

		Time (Sec)			
		.00	22.20(1)	157.19(2)	1500.00(3)
		Energy (Million Btu)			
Initial Energy	In RCS, Acc, SG	667.41	667.41	667.41	667.41
Added Energy	Pumped Injection	.00	.00	4.32	47.43
	Decay Heat	.00	6.28	22.91	124.26
	Heat From Secondary	.00	-.20	-.20	-.20
	Total Added	.00	6.08	27.03	171.49
Total Available		667.41	673.49	694.44	838.90
Distribution	Reactor Coolant	247.64	17.46	25.13	25.76
	Accumulator	14.95	10.29	.00	.00
	Core Stored	22.89	9.64	1.05	1.05
	Vessel Thin Metal	15.13	12.85	3.03	3.03
	Vessel Thick Metal	27.45	27.26	22.47	13.67
	RCS Loop Plus Steam Generators	339.34	332.86	325.18	323.60
	Total Contents	667.41	410.38	376.87	367.12
Effluent	Break Flow	.00	262.62	312.35	431.35
	ECCS Spill	.00	.00	4.73	39.94
	Total Effluent	.00	262.62	317.08	471.29
Total Accountable		667.41	673.00	693.95	838.41

(1) End of Blowdown
(2) End of Reflood
(3) End of Injection Phase

Table 3.1-25
Double-Ended Hot Leg Break Energy Balance Maximum Safeguards (BVPS-1 Case 2L)

		Time (Sec)			
		.00	22.20 (1)	129.55 (2)	1500.00 (3)
		Energy (Million Btu)			
Initial Energy	In RCS, Acc, SG	667.41	667.41	667.41	667.41
Added Energy	Pumped Injection	.00	.00	5.52	76.17
	Decay Heat	.00	6.28	19.90	124.26
	Heat From Secondary	.00	-.20	-.20	-.20
	Total Added	.00	6.08	25.22	200.22
Total Available		667.41	673.49	692.63	867.63
Distribution	Reactor Coolant	247.64	17.46	25.09	25.63
	Accumulator	14.95	10.29	.00	.00
	Core Stored	22.89	9.64	1.05	1.05
	Vessel Thin Metal	15.13	12.85	3.03	3.03
	Vessel Thick Metal	27.45	27.26	23.29	13.67
	RCS Loop Plus Steam Generators	339.34	332.86	324.58	323.07
	Total Contents	667.41	410.38	377.05	366.47
Effluent	Break Flow	.00	262.62	309.72	432.75
	ECCS Spill	.00	.00	5.36	67.93
	Total Effluent	.00	262.62	315.09	500.68
Total Accountable		667.41	673.00	692.14	867.14

(1) End of Blowdown
(2) End of Reflood
(3) End of Injection Phase

Table 3.1-26
Double-Ended Hot Leg Break - Minimum Safeguards
Principle Parameters During Reflood (BVPS-1 Case 1L)

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
.00	292.30	.000	.000	.00	.00	.333	.0	.0	.0	73.00
.21	286.56	69.646	.000	.66	-.12	.999	115.8	108.7	.0	74.40
.27	283.29	84.134	.000	1.04	-.50	.829	115.5	108.4	.0	74.40
.48	279.37	5.587	.364	1.50	-.46	.885	113.6	106.6	.0	74.40
.61	279.09	5.496	.407	1.54	.05	.889	112.9	105.9	.0	74.40
2.88	273.20	7.810	.738	2.00	8.11	.920	101.3	94.5	.0	74.39
6.01	262.52	9.788	.827	2.49	15.63	.924	88.5	82.0	54.6	74.38
6.12	262.16	9.765	.829	2.50	15.63	.924	88.2	81.7	54.3	74.38
10.08	251.44	9.232	.847	3.00	15.63	.924	78.8	72.3	46.8	74.37
14.51	243.00	8.870	.852	3.50	15.63	.923	70.6	64.0	39.8	74.35
19.16	236.43	8.552	.851	4.00	15.63	.923	63.6	57.0	34.0	74.34
23.93	231.08	8.246	.849	4.50	15.63	.923	57.7	51.1	29.1	74.32
28.81	226.45	7.941	.846	5.00	15.63	.922	52.6	45.9	25.1	74.30
33.77	222.27	7.634	.843	5.50	15.63	.922	48.1	41.4	21.7	74.28
34.01	222.08	7.619	.843	5.52	15.63	.922	47.9	41.2	21.5	74.28
35.45	220.94	7.533	.842	5.67	15.62	.922	6.7	.0	.0	73.00
39.09	218.74	5.799	.826	6.00	13.12	.916	6.8	.0	.0	73.00

Table 3.1-26 (Continued)
Double-Ended Hot Leg Break - Minimum Safeguards
Principle Parameters During Reflood (BVPS-1 Case 1L)

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
46.03	217.47	3.625	.795	6.50	10.50	.900	7.0	.0	.0	73.00
55.52	218.73	2.357	.766	7.00	9.33	.880	7.1	.0	.0	73.00
67.48	221.59	1.899	.753	7.50	9.28	.868	7.1	.0	.0	73.00
80.62	224.67	1.796	.750	8.00	9.68	.865	7.1	.0	.0	73.00
94.11	227.24	1.778	.751	8.50	10.20	.865	7.1	.0	.0	73.00
105.01	228.75	1.777	.752	8.90	10.62	.865	7.1	.0	.0	73.00
107.70	229.04	1.777	.752	9.00	10.73	.866	7.1	.0	.0	73.00
121.34	230.03	1.778	.753	9.50	11.26	.866	7.1	.0	.0	73.00
134.99	230.23	1.779	.753	10.00	11.80	.866	7.1	.0	.0	73.00

Table 3.1-27
Double-Ended Hot Leg Break - Maximum Safeguards
Principle Parameters During Reflood (BVPS-1 Case 2L)

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
.00	292.30	.000	.000	.00	.00	.333	.0	.0	.0	73.00
.20	286.90	68.926	.000	.62	-.07	.999	120.2	108.7	.0	74.35
.27	283.03	86.548	.001	1.07	-.51	.684	119.9	108.4	.0	74.35
.46	279.34	5.943	.365	1.50	-.48	.889	118.0	106.6	.0	74.35
.61	278.99	5.736	.416	1.55	.14	.892	117.2	105.8	.0	74.35
2.80	273.02	7.987	.739	2.00	8.26	.920	105.8	94.7	.0	74.33
5.01	265.28	9.928	.816	2.36	14.71	.924	95.8	85.0	.0	74.32
5.99	261.87	9.769	.828	2.50	15.63	.924	92.9	82.1	59.1	74.32
9.95	251.12	9.235	.847	3.00	15.63	.924	83.5	72.6	51.5	74.30
14.37	242.70	8.873	.851	3.50	15.63	.923	75.2	64.3	44.5	74.27
19.01	236.16	8.555	.851	4.00	15.63	.923	68.2	57.3	38.6	74.25
23.78	230.84	8.248	.849	4.50	15.63	.923	62.2	51.3	33.7	74.23
28.65	226.26	7.943	.846	5.00	15.63	.922	57.1	46.1	29.6	74.20
33.62	222.11	7.635	.843	5.50	15.63	.922	52.6	41.6	26.2	74.18
34.01	221.80	7.610	.843	5.54	15.63	.922	52.3	41.2	25.9	74.18
35.45	220.67	7.524	.842	5.68	15.62	.922	11.1	.0	.0	73.00
38.84	218.47	6.270	.831	6.00	13.79	.918	11.2	.0	.0	73.00

Table 3.1-27 (Continued)
Double-Ended Hot Leg Break - Maximum Safeguards
Principle Parameters During Reflood (BVPS-1 Case 2L)

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
45.08	216.34	4.640	.811	6.50	11.86	.909	11.3	.0	.0	73.00
52.53	215.66	3.653	.795	7.00	10.98	.900	11.4	.0	.0	73.00
60.99	215.82	3.182	.786	7.50	10.86	.895	11.4	.0	.0	73.00
70.01	216.17	2.997	.782	8.00	11.14	.892	11.4	.0	.0	73.00
79.27	216.28	2.932	.781	8.50	11.59	.891	11.4	.0	.0	73.00
88.62	215.94	2.911	.780	9.00	12.10	.891	11.4	.0	.0	73.00
97.99	215.09	2.904	.779	9.50	12.63	.891	11.4	.0	.0	73.00
107.35	213.73	2.901	.779	10.00	13.16	.891	11.4	.0	.0	73.00

Table 3.1-28
System Parameters Initial Conditions For Thermal Uprate (BVPS-2)

Parameters	Value
Core Thermal Power (MWt)*	2917.4
Reactor Coolant System Total Flowrate (lbm/sec)	27583.3
Vessel Outlet Temperature (°F)*	621
Core Inlet Temperature (°F)*	547.1
Vessel Average Temperature (°F)*	584
Initial Steam Generator Steam Pressure (psia)	826
Steam Generator Design	51
Steam Generator Tube Plugging (percent)	0
Initial Steam Generator Secondary Side Mass (lbm)*	127881
Assumed Maximum Containment Backpressure (psia)	59.7
Accumulator	
Water Volume (ft^3) per accumulator	1127.8
N_2 Cover Gas Pressure (psia)	575
Temperature (°F)	105

Note:
* The Core Power, RCS Temperatures, and Secondary Side Mass values listed above include uncertainty allowance.

Table 3.1-29
Safety Injection Flow Minimum Safeguards (BVPS-2)

RCS Pressure (psig)	Total Flow (GPM)
Injection Mode (Reflood Phase)	
0	4254.9
20	3895.7
50	3264.7
95	1679.6
100	1412.5
150	388.8
200	383.6
400	362.8
600	341.8
Cold Leg Recirculation Mode	
0	3767

Note:

A maximum Safety Injection Temperature of 65°F was used during the Injection Phase and 120°F was used during the Recirculation Phase.

Table 3.1-30
Safety Injection Flow Maximum Safeguards (BVPS-2)

RCS Pressure (psig)	Total Flow (GPM)
Injection Mode (Reflood Phase)	
0	6148.5
20	5696.8
50	5019.3
100	3265.5
130	1481.0
150	847.9
200	840.0
400	803.7
600	771.0
Cold Leg Recirculation Mode	
0	6228.6

Note:

A maximum Safety Injection Temperature of 65°F was used during the Injection Phase. A maximum recirculation temperature of 120°F was used during the Recirculation Phase for the CIB failure case (Case 2L) and 150°F was used for the SW failure case (Case $2L_1$).

Table 3.1-31
Double-Ended Hot-Leg Break Blowdown Mass and Energy Releases (BVPS-2) (Case 3L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
.00000	.0	.0	.0	.0
.00110	46506.1	29844.9	46504.5	29843.1
.00214	46366.6	29754.2	46133.9	29599.7
.101	41311.9	26879.3	26481.2	16957.0
.201	34980.7	22842.8	23912.4	15242.1
.301	34192.1	22255.7	21041.5	13262.6
.401	33317.1	21650.8	19607.0	12161.5
.502	32470.5	21091.0	18714.2	11411.2
.602	32391.2	21031.5	18148.0	10894.0
.701	32410.5	21055.5	17608.8	10426.7
.801	32014.1	20841.8	17255.8	10094.8
.902	31474.9	20557.3	16960.8	9818.1
1.00	31197.4	20470.2	16684.8	9571.9
1.10	30928.9	20413.4	16497.6	9391.0
1.20	30573.5	20308.0	16344.4	9239.6
1.30	30169.3	20163.3	16252.2	9131.9
1.40	29740.4	20003.3	16213.2	9060.6
1.50	29287.3	19827.8	16220.9	9021.2
1.60	28761.6	19606.8	16261.8	9003.7
1.70	28147.8	19322.1	16325.4	9002.8
1.80	27476.6	18995.0	16405.4	9013.9
1.90	26804.3	18663.2	16493.0	9032.5
2.00	26155.8	18342.7	16579.9	9054.3
2.10	25493.2	18003.5	16664.1	9077.9
2.20	24793.6	17624.1	16742.3	9101.3
2.30	24103.8	17236.2	16809.8	9122.1
2.40	23435.8	16850.3	16867.0	9140.0
2.50	22788.1	16467.0	16913.2	9154.2

Table 3.1-31 (Continued)
Double-Ended Hot-Leg Break Blowdown Mass and Energy Releases (BVPS-2) (Case 3L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
2.60	22157.1	16083.7	16946.5	9163.7
2.70	21562.7	15712.5	16966.1	9167.6
2.80	20999.5	15344.6	16971.5	9165.5
2.90	20499.3	15010.2	16963.7	9157.6
3.00	20034.1	14686.4	16942.7	9143.7
3.10	19628.5	14391.5	16909.2	9124.1
3.20	19285.5	14129.6	16865.5	9099.9
3.30	18980.9	13882.7	16811.7	9070.9
3.40	18721.7	13659.4	16748.3	9037.4
3.50	18506.3	13459.3	16677.2	9000.3
3.60	18320.5	13273.7	16597.5	8959.0
3.70	18171.4	13109.6	16511.1	8914.5
3.80	18047.1	12959.4	16417.7	8866.6
3.90	17946.8	12824.1	16317.5	8815.3
4.00	17867.0	12702.5	16211.5	8761.2
4.20	17776.0	12507.4	15983.3	8645.0
4.40	17808.4	12393.5	15726.1	8514.1
4.60	17927.7	12343.5	15429.9	8363.2
4.80	18115.9	12329.6	15074.2	8181.1
5.00	18396.0	12352.9	14740.5	8012.5
5.20	18767.1	12418.9	14296.3	7783.7
5.40	19352.6	12607.5	13848.4	7553.6
5.60	14305.2	10429.7	13391.3	7319.0
5.80	14241.6	10255.5	12931.5	7083.4
6.00	14364.3	10266.5	12515.2	6872.0
6.20	14439.6	10292.3	12097.5	6659.0
6.40	14689.4	10301.9	11637.4	6421.2
6.60	15049.1	10399.0	11192.0	6191.2

Table 3.1-31 (Continued)
Double-Ended Hot-Leg Break Blowdown Mass and Energy Releases (BVPS-2) (Case 3L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
6.80	14999.6	10249.0	10769.6	5973.6
7.00	15365.5	10330.1	10353.6	5758.9
7.20	15683.6	10402.0	9948.6	5549.8
7.40	15971.8	10469.1	9571.7	5355.7
7.60	16258.9	10543.6	9214.4	5171.8
7.80	16594.6	10651.0	8875.1	4997.0
8.00	17158.0	10907.0	8556.8	4833.1
8.20	17053.0	10792.2	8246.6	4673.3
8.40	16725.6	10538.7	7949.2	4520.3
8.60	14694.9	9424.6	7660.0	4371.6
8.80	13878.9	8964.1	7380.7	4228.8
9.00	13826.1	8916.4	7115.7	4094.4
9.20	13789.1	8883.5	6866.0	3969.2
9.40	13740.1	8844.0	6639.3	3856.8
9.60	13672.9	8787.6	6416.5	3745.9
9.80	13461.2	8649.8	6204.1	3640.7
10.0	12878.5	8314.2	5996.5	3538.1
10.2	12189.5	7927.5	5796.5	3440.0
10.4	11803.9	7705.9	5599.5	3344.4
10.4	11799.7	7703.4	5596.6	3342.9
10.4	11796.0	7701.2	5594.1	3341.7
10.6	11564.5	7566.0	5411.4	3254.3
10.8	11330.9	7432.4	5230.6	3168.7
11.0	11053.8	7277.7	5055.7	3086.6
11.2	10712.5	7091.0	4886.8	3007.9
11.4	10331.8	6886.4	4722.8	2931.8
11.6	9964.0	6691.8	4566.3	2859.7
11.8	9625.7	6515.9	4413.7	2789.9

Table 3.1-31 (Continued)
Double-Ended Hot-Leg Break Blowdown Mass and Energy Releases (BVPS-2) (Case 3L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
12.0	9308.6	6355.2	4265.5	2722.9
12.2	8992.6	6198.7	4122.2	2658.5
12.4	8650.4	6032.5	3981.1	2595.1
12.6	8260.6	5847.8	3839.5	2531.7
12.8	7824.8	5648.0	3689.7	2464.3
13.0	7355.1	5441.4	3528.5	2392.9
13.2	6851.5	5229.8	3350.6	2316.8
13.4	6334.7	5023.8	3157.0	2236.8
13.6	5794.9	4815.8	2947.5	2153.0
13.8	5241.0	4606.0	2732.2	2068.6
14.0	4683.9	4393.7	2518.7	1983.8
14.2	4106.9	4114.4	2321.9	1903.4
14.4	3655.6	3782.3	2145.6	1826.3
14.6	3390.8	3538.3	1997.8	1757.7
14.8	3204.0	3361.8	1873.2	1695.8
15.0	3043.7	3210.9	1771.0	1645.7
15.2	2853.9	3053.5	1677.5	1598.2
15.4	2616.4	2882.7	1596.2	1554.9
15.6	2347.8	2697.4	1518.1	1516.8
15.8	2093.8	2499.3	1435.1	1485.2
16.0	1893.8	2304.3	1350.3	1450.4
16.2	1707.1	2093.3	1263.9	1416.2
16.4	1547.9	1912.1	1182.6	1368.2
16.6	1412.2	1757.7	1132.2	1346.8
16.8	1340.1	1676.1	1022.5	1239.5
17.0	1261.1	1583.5	953.6	1166.7
17.2	1162.6	1461.4	899.6	1105.5
17.4	1067.4	1343.3	858.0	1056.8

Table 3.1-31 (Continued)
Double-Ended Hot-Leg Break Blowdown Mass and Energy Releases (BVPS-2) (Case 3L)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
17.6	998.4	1256.2	811.0	1000.7
17.8	915.2	1153.4	726.6	898.5
18.0	836.1	1056.2	630.5	782.0
18.2	758.4	959.6	562.2	698.7
18.4	664.0	841.2	523.1	651.9
18.6	593.8	753.6	405.0	503.7
18.7	559.5	710.5	369.6	461.6
18.8	317.2	402.1	344.0	430.2
19.0	.0	.0	160.9	201.8
19.2	.0	.0	.0	.0

* Mass and Energy exiting from the RV side of the break

** Mass and Energy exiting from the SG side of the break

Table 3.1-32
Double-Ended Pump Suction Break Blowdown Mass and Energy Releases
(Same for all DEPS Runs) (BVPS-2) (Case 1L, Case 2L & Case 2L1)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
.00000	.0	.0	.0	.0
.00109	88813.2	47922.9	40351.7	21722.1
.101	40209.7	21715.8	20435.8	10994.1
.202	41011.2	22298.7	22609.3	12171.8
.301	44814.9	24586.6	23088.2	12441.9
.401	45435.3	25200.1	22751.7	12274.7
.501	44833.6	25178.3	22031.3	11896.1
.602	44232.4	25156.0	21364.1	11542.5
.702	44540.0	25626.6	20891.4	11291.2
.801	44177.2	25680.9	20541.8	11106.2
.901	43156.3	25326.6	20278.1	10967.8
1.00	41991.4	24872.3	20086.3	10867.0
1.10	40887.0	24443.5	19955.7	10798.7
1.20	39821.3	24030.8	19897.4	10768.6
1.30	38772.9	23621.4	19905.5	10774.0
1.40	37783.2	23230.6	19936.5	10791.0
1.50	36908.7	22887.7	19940.9	10792.8
1.60	36142.5	22591.2	19917.5	10778.9
1.70	35446.0	22327.1	19885.1	10759.9
1.80	34731.2	22053.6	19850.2	10739.8
1.90	33965.9	21761.5	19793.1	10707.8
2.00	33145.0	21449.3	19693.9	10653.2
2.10	32124.1	21021.3	19534.0	10565.6
2.20	30989.8	20520.6	19217.8	10393.1
2.30	29718.8	19926.4	18898.7	10220.1
2.40	28370.0	19265.1	18585.8	10050.1
2.50	26271.4	18046.5	18237.6	9861.0

Table 3.1-32 (Continued)
Double-Ended Pump Suction Break Blowdown Mass and Energy Releases
(Same for all DEPS Runs) (BVPS-2) (Case 1L, Case 2L & Case 2L1)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
2.60	22526.4	15619.5	17936.3	9698.1
2.70	20341.8	14248.6	17630.1	9533.0
2.80	19301.7	13612.7	17291.8	9350.7
2.90	18053.8	12758.2	16997.9	9193.2
3.00	17127.4	12122.9	16751.9	9062.3
3.10	16396.6	11622.5	16516.2	8937.0
3.20	15708.0	11149.0	16291.8	8818.0
3.30	15096.6	10734.6	16088.7	8710.8
3.40	14548.9	10370.0	15901.7	8612.5
3.50	14063.1	10050.5	15727.2	8520.8
3.60	13636.3	9772.8	15539.9	8422.0
3.70	13269.5	9536.7	15426.3	8364.0
3.80	12948.4	9329.5	15291.4	8293.5
3.90	12609.6	9105.9	15123.9	8205.2
4.00	12288.5	8895.2	14986.8	8133.8
4.20	11754.8	8549.6	14723.3	7996.4
4.40	11324.6	8258.2	14466.5	7862.6
4.60	10987.7	8022.4	14240.5	7745.6
4.80	10702.1	7816.8	14000.8	7621.4
5.00	10461.2	7642.5	13753.4	7492.8
5.20	10236.1	7478.4	13395.5	7303.5
5.40	10103.0	7378.3	13027.1	7109.3
5.60	10296.5	7512.6	14648.7	8004.6
5.80	10519.3	7740.8	14501.0	7929.5
6.00	9862.2	7865.5	14373.1	7867.2
6.20	8477.2	7510.2	14106.8	7727.0
6.40	7876.2	7245.0	13799.5	7564.8

Table 3.1-32 (Continued)
Double-Ended Pump Suction Break Blowdown Mass and Energy Releases
(Same for all DEPS Runs) (BVPS-2) (Case 1L, Case 2L & Case 2L1)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
6.60	7591.4	7038.1	13470.1	7390.4
6.80	7492.7	6810.1	13184.9	7239.5
7.00	7587.5	6652.4	12948.4	7113.8
7.20	7790.2	6576.2	12740.3	6999.6
7.40	7953.2	6499.0	12637.0	6938.7
7.60	8022.6	6416.0	12505.7	6858.8
7.80	8024.0	6352.5	12337.7	6758.3
8.00	7926.6	6244.8	12148.7	6647.7
8.20	7812.6	6139.8	12007.4	6565.3
8.40	7690.2	6041.2	11864.5	6482.9
8.60	7561.1	5950.0	11693.4	6385.2
8.80	7428.3	5864.5	11518.0	6285.8
9.00	7289.7	5779.9	11351.7	6192.3
9.20	7145.6	5694.1	11181.4	6097.4
9.40	6996.8	5606.2	11002.6	5998.2
9.60	6847.7	5517.9	10828.1	5901.7
9.80	6698.6	5429.1	10659.4	5808.7
10.0	6555.3	5346.1	10488.0	5714.3
10.2	6411.9	5258.8	10311.2	5617.0
10.4	6275.6	5171.5	10141.4	5524.2
10.6	6144.5	5084.6	9972.5	5432.1
10.8	6017.4	4999.1	9805.3	5340.9
11.0	5892.8	4915.0	9643.4	5252.7
11.2	5771.6	4833.4	9483.7	5165.6
11.4	5647.4	4750.4	9325.8	5079.5
11.6	5513.4	4661.3	9141.1	4978.6
11.8	5368.9	4563.2	8967.0	4884.5

Table 3.1-32 (Continued)
Double-Ended Pump Suction Break Blowdown Mass and Energy Releases
(Same for all DEPS Runs) (BVPS-2) (Case 1L, Case 2L & Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
12.0	5226.4	4458.3	8806.8	4798.4
12.2	5093.0	4350.7	8629.6	4702.5
12.4	4969.0	4243.7	8462.8	4612.5
12.6	4846.9	4136.6	8287.8	4518.0
12.8	4723.7	4028.9	8117.7	4426.6
13.0	4602.9	3923.7	7878.7	4296.7
13.2	4490.6	3826.3	7607.2	4150.3
13.4	4390.7	3739.2	7478.0	4077.9
13.6	4297.3	3660.8	7252.9	3927.2
13.8	4205.1	3592.2	7209.6	3851.5
14.0	4112.3	3533.8	7192.1	3772.8
14.2	4017.5	3484.4	7146.4	3671.8
14.4	3913.0	3439.8	6964.3	3502.6
14.6	3800.2	3402.0	6677.3	3287.1
14.8	3675.5	3368.9	6392.0	3081.2
15.0	3543.1	3342.6	6195.1	2929.4
15.2	3403.2	3321.2	5974.9	2779.0
15.4	3253.9	3306.0	5754.2	2635.7
15.6	3057.1	3259.2	5532.0	2497.5
15.8	2763.2	3138.5	5062.7	2251.5
16.0	2468.0	2958.4	4680.7	2042.7
16.2	2250.5	2759.1	4428.1	1892.9
16.4	2048.5	2529.8	4163.1	1746.1
16.6	1874.5	2323.3	3778.8	1555.6
16.8	1723.6	2142.6	3381.0	1362.8
17.0	1594.6	1986.4	3022.3	1189.2
17.2	1469.3	1833.8	2783.7	1067.8

Table 3.1-32 (Continued)
Double-Ended Pump Suction Break Blowdown Mass and Energy Releases
(Same for all DEPS Runs) (BVPS-2) (Case 1L, Case 2L & Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
17.4	1352.3	1690.7	2650.6	992.2
17.6	1228.0	1538.0	2602.1	952.7
17.8	1122.4	1407.9	2570.5	921.5
18.0	1022.8	1284.4	2615.4	915.7
18.2	911.4	1146.5	2760.6	941.6
18.4	809.0	1019.0	2941.7	976.9
18.6	704.7	888.5	3034.8	983.8
18.8	603.4	761.4	3003.2	952.9
19.0	509.2	643.1	2786.6	867.7
19.2	424.2	536.2	2538.9	778.1
19.4	349.8	442.5	2287.9	691.3
19.6	287.7	364.2	2025.4	604.1
19.8	250.1	316.8	1762.7	519.4
20.0	199.4	252.7	1510.9	440.4
20.2	125.5	159.4	1244.0	359.1
20.4	48.7	62.0	966.5	276.8
20.6	70.3	89.8	709.7	202.1
20.8	.0	.0	462.7	131.4
21.0	.0	.0	214.2	60.9
21.2	.0	.0	32.0	9.1
21.4	.0	.0	.0	.0

* Mass and Energy exiting the SG side of the break

** Mass and Energy exiting the pump side of the break

Table 3.1-33
Double-Ended Pump Suction Break Minimum Safeguards
Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
21.4	.0	.0	.0	.0
21.9	.0	.0	.0	.0
22.1	.0	.0	.0	.0
22.2	.0	.0	.0	.0
22.3	.0	.0	.0	.0
22.4	.0	.0	.0	.0
22.4	.0	.0	.0	.0
22.5	57.4	67.5	.0	.0
22.6	20.7	24.4	.0	.0
22.7	18.2	21.4	.0	.0
22.9	21.6	25.4	.0	.0
23.0	31.8	37.4	.0	.0
23.1	37.0	43.6	.0	.0
23.2	43.0	50.6	.0	.0
23.3	48.3	56.8	.0	.0
23.4	53.7	63.3	.0	.0
23.5	58.3	68.6	.0	.0
23.6	61.7	72.7	.0	.0
23.7	65.1	76.7	.0	.0
23.8	66.8	78.7	.0	.0
23.8	68.4	80.6	.0	.0
23.9	71.6	84.3	.0	.0
24.0	74.7	87.9	.0	.0
24.1	77.6	91.4	.0	.0
24.2	80.5	94.8	.0	.0
24.3	83.3	98.1	.0	.0
24.4	86.0	101.3	.0	.0

Table 3.1-33 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards
Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
25.4	109.9	129.5	.0	.0
26.4	129.7	152.9	.0	.0
27.4	155.3	183.0	551.6	47.2
28.4	372.2	440.4	3820.8	431.3
28.5	374.6	443.4	3840.0	436.1
29.5	374.3	443.0	3826.9	440.5
30.5	367.7	435.1	3756.9	434.8
31.5	360.8	426.8	3683.8	428.3
32.5	353.9	418.7	3610.7	421.8
33.2	349.2	413.1	3560.3	417.2
33.5	347.3	410.7	3538.9	415.3
34.5	340.8	403.1	3468.9	408.9
35.5	334.6	395.7	3400.9	402.6
36.5	328.7	388.6	3335.0	396.5
37.5	322.9	381.8	3271.1	390.6
38.5	317.4	375.3	3209.3	384.8
39.5	312.2	369.0	3149.4	379.3
40.5	307.1	363.0	3091.4	373.9
41.5	302.2	357.1	3035.2	368.6
42.5	297.5	351.6	2980.7	363.5
43.5	293.0	346.2	2927.8	358.5
44.5	288.6	341.0	2876.4	353.7
45.5	284.4	336.0	2826.4	349.0
46.5	280.3	331.1	2777.8	344.4
46.9	278.7	329.2	2758.7	342.6
47.5	276.4	326.4	2730.5	339.9
48.5	272.5	321.9	2684.4	335.6

Table 3.1-33 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards
Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
49.5	268.8	317.5	2639.5	331.3
50.5	265.2	313.2	2595.7	327.2
51.5	261.8	309.1	2553.0	323.1
52.5	258.4	305.1	2511.2	319.1
53.5	255.1	301.2	2470.4	315.2
54.5	251.9	297.4	2430.5	311.4
55.1	250.0	295.2	2407.0	309.2
55.5	248.8	293.7	2391.5	307.7
56.5	245.8	290.2	2353.3	304.0
57.5	242.8	286.7	2315.9	300.4
58.5	240.0	283.3	2279.2	296.9
59.5	237.2	279.9	2243.3	293.4
60.5	234.4	276.7	2208.0	290.0
61.5	231.8	273.5	2173.4	286.6
62.5	229.1	270.4	2139.5	283.3
63.5	218.0	257.2	232.5	100.6
64.5	239.8	283.0	238.2	111.6
65.5	236.1	278.7	237.1	109.8
66.5	232.4	274.3	235.9	107.9
67.5	228.7	269.9	234.7	106.0
68.5	225.1	265.6	233.5	104.2
69.5	221.5	261.4	232.4	102.5
70.5	218.1	257.3	231.3	100.8
71.5	214.6	253.2	230.3	99.1
72.5	211.2	249.2	229.2	97.5
73.3	208.5	246.0	228.4	96.2
73.5	207.8	245.2	228.1	95.8

Table 3.1-33 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards
Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
74.5	204.6	241.4	227.1	94.3
75.5	201.8	238.1	226.1	93.0
76.5	199.0	234.8	225.2	91.6
77.5	196.3	231.5	224.2	90.4
78.5	193.6	228.4	223.3	89.1
79.5	191.0	225.3	222.4	87.9
80.5	188.5	222.3	221.6	86.7
81.5	185.9	219.3	220.7	85.6
82.5	183.5	216.4	219.9	84.5
84.5	178.8	210.8	218.3	82.3
86.5	174.2	205.5	216.8	80.3
88.5	169.9	200.4	215.4	78.4
90.5	165.9	195.6	214.1	76.6
92.5	162.0	191.0	212.9	74.9
94.5	158.3	186.6	211.7	73.3
95.0	157.4	185.6	211.4	73.0
96.5	154.9	182.6	210.6	71.9
98.5	151.6	178.7	209.6	70.5
100.5	148.5	175.1	208.6	69.2
102.5	145.6	171.6	207.7	68.0
104.5	142.9	168.4	206.9	66.9
106.5	140.4	165.4	206.1	65.8
108.5	138.0	162.6	205.4	64.8
110.5	135.8	160.0	204.7	64.0
112.5	133.7	157.6	204.1	63.1
114.5	131.8	155.4	203.5	62.4
116.5	130.1	153.3	203.0	61.7

Table 3.1-33 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards
Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
118.5	128.5	151.4	202.5	61.0
120.5	127.0	149.6	202.1	60.4
120.8	126.8	149.4	202.0	60.3
122.5	125.6	148.0	201.7	59.9
124.5	124.4	146.6	201.3	59.4
126.5	123.2	145.2	201.0	58.9
128.5	122.2	144.0	200.7	58.5
130.5	121.3	142.9	200.4	58.1
132.5	120.4	141.9	200.1	57.8
134.5	119.7	141.0	199.9	57.5
136.5	119.0	140.2	199.7	57.2
138.5	118.4	139.5	199.5	57.0
140.5	117.9	138.9	199.3	56.7
142.5	117.4	138.3	199.2	56.5
144.5	117.0	137.8	199.0	56.4
146.5	116.6	137.4	198.9	56.2
148.5	116.3	137.1	198.8	56.1
149.8	116.2	136.9	198.8	56.0
150.5	116.1	136.8	198.7	55.9
152.5	115.9	136.5	198.6	55.8
154.5	116.0	136.7	198.7	55.9
156.5	116.2	136.9	199.3	56.0
158.5	116.3	137.0	200.2	56.3
160.5	116.3	137.1	201.7	56.6
162.5	116.4	137.1	203.5	57.0
164.5	116.4	137.2	205.7	57.5
166.5	116.4	137.1	208.2	58.1
168.5	116.3	137.0	211.0	58.7

Table 3.1-33 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards
Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
170.5	116.1	136.8	214.1	59.3
172.5	115.8	136.5	217.4	60.0
174.5	115.5	136.0	220.9	60.7
176.5	115.0	135.4	224.6	61.4
178.5	114.3	134.7	228.4	62.2
180.5	113.6	133.8	232.4	62.9
180.6	113.5	133.7	232.6	62.9
182.5	112.6	132.7	236.6	63.7
184.5	111.6	131.5	241.0	64.4
186.5	110.4	130.0	245.4	65.2
188.5	109.0	128.4	250.1	66.0
190.5	107.4	126.5	254.9	66.8
192.5	107.0	126.0	258.2	67.2
194.5	106.7	125.7	261.2	67.5
196.5	106.4	125.4	263.8	67.8
198.5	106.1	125.0	266.3	68.0
200.5	105.7	124.6	268.7	68.1
202.5	105.4	124.1	270.8	68.1
204.5	104.9	123.6	272.8	68.1
206.5	104.5	123.1	274.7	68.1
208.5	104.1	122.6	276.4	68.0
210.5	103.6	122.0	278.1	67.9
212.5	103.1	121.5	279.7	67.7
214.2	102.7	121.0	280.9	67.6

* Mass and Energy exiting the SG side of the break
** Mass and Energy exiting the pump side of the break

Table 3.1-34
Double-Ended Pump Suction Break - Minimum Safeguards Principle Parameters During Reflood (BVPS-2) (Case 1L)

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
21.4	169.0	.000	.000	.00	.00	.333	.0	.0	.0	.00
22.2	166.4	22.694	.000	.67	1.40	.000	6117.9	6117.9	.0	74.50
22.4	165.1	24.146	.000	1.07	1.32	.000	6067.6	6067.6	.0	74.50
22.7	164.5	2.530	.113	1.32	1.96	.298	5952.2	5952.2	.0	74.50
22.9	164.4	2.633	.130	1.35	2.40	.310	5934.5	5934.5	.0	74.50
23.0	164.4	2.595	.163	1.37	2.69	.361	5887.9	5887.9	.0	74.50
23.2	164.4	2.604	.205	1.41	3.34	.390	5842.1	5842.1	.0	74.50
23.8	164.4	2.494	.302	1.50	4.99	.427	5710.0	5710.0	.0	74.50
24.4	164.5	2.425	.386	1.59	6.83	.442	5574.6	5574.6	.0	74.50
28.4	164.9	4.268	.635	2.01	15.61	.663	4786.7	4404.9	.0	71.19
30.5	165.1	3.918	.684	2.25	15.62	.658	4491.7	4109.2	.0	70.97
33.2	165.6	3.627	.710	2.51	15.62	.651	4221.1	3828.9	.0	70.64
39.5	167.5	3.232	.731	3.00	15.62	.635	3718.3	3307.3	.0	69.91
46.9	170.4	2.942	.739	3.50	15.62	.617	3263.8	2837.3	.0	69.08
55.1	174.1	2.711	.741	4.00	15.62	.599	2862.7	2423.8	.0	68.14
62.5	177.6	2.547	.741	4.42	15.62	.584	2560.4	2113.3	.0	67.25
63.5	178.1	2.534	.741	4.48	15.61	.578	454.2	.0	.0	33.00
64.5	178.6	2.631	.744	4.53	15.49	.589	442.9	.0	.0	33.00

Table 3.1-34 (Continued)
Double-Ended Pump Suction Break - Minimum Safeguards Principle Parameters During Reflood (BVPS-2) (Case 1L)

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
73.3	183.8	2.357	.741	5.00	14.63	.574	452.9	.0	.0	33.00
84.5	191.7	2.098	.738	5.54	13.96	.555	458.3	.0	.0	33.00
95.0	199.7	1.912	.737	6.00	13.65	.537	461.5	.0	.0	33.00
108.5	210.1	1.743	.736	6.54	13.57	.518	464.1	.0	.0	33.00
120.8	218.6	1.644	.736	7.00	13.73	.504	465.5	.0	.0	33.00
136.5	227.9	1.572	.738	7.55	14.10	.493	466.5	.0	.0	33.00
149.8	234.7	1.541	.741	8.00	14.51	.490	466.8	.0	.0	33.00
152.5	236.0	1.537	.742	8.09	14.60	.489	466.9	.0	.0	33.00
154.5	236.9	1.537	.742	8.16	14.67	.489	466.9	.0	.0	33.00
166.5	242.2	1.531	.745	8.55	15.04	.491	466.8	.0	.0	33.00
180.6	247.9	1.496	.748	9.00	15.37	.488	467.2	.0	.0	33.00
198.5	254.0	1.419	.750	9.55	15.57	.480	467.8	.0	.0	33.00
214.2	258.7	1.364	.752	10.00	15.61	.482	467.8	.0	.0	33.00

Table 3.1-35
Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
214.2	121.2	152.6	347.1	84.9
219.2	121.9	153.4	346.5	84.6
224.2	121.5	153.0	346.8	84.5
229.2	121.2	152.6	347.1	84.4
234.2	120.8	152.2	347.5	84.3
239.2	121.5	153.0	346.8	84.0
244.2	121.1	152.5	347.2	83.9
249.2	120.8	152.1	347.5	83.8
254.2	120.4	151.7	347.9	83.7
259.2	121.1	152.5	347.2	83.4
264.2	120.7	152.0	347.6	83.3
269.2	120.4	151.6	347.9	83.2
274.2	120.0	151.1	348.3	86.0
279.2	120.7	151.9	347.7	85.6
284.2	120.3	151.5	348.0	85.6
289.2	119.9	151.0	348.4	85.5
294.2	120.6	151.8	347.7	85.1
299.2	120.2	151.4	348.1	85.0
304.2	119.8	150.9	348.5	84.9
309.2	119.5	150.5	348.8	84.8
314.2	120.1	151.2	348.2	84.5
319.2	119.7	150.8	348.6	84.4
324.2	119.4	150.3	349.0	84.3
329.2	119.9	151.0	348.4	83.9
334.2	119.6	150.6	348.7	83.9
339.2	119.2	150.1	349.1	83.8
344.2	118.8	149.6	349.5	83.7

Table 3.1-35 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
349.2	119.4	150.4	348.9	83.3
354.2	119.0	149.9	349.3	83.2
359.2	118.7	149.4	349.6	83.1
364.2	119.2	150.1	349.1	82.8
369.2	118.8	149.6	349.5	82.7
374.2	118.5	149.2	349.8	82.6
379.2	119.0	149.9	349.3	82.3
384.2	118.6	149.4	349.7	82.2
389.2	118.2	148.9	350.1	82.1
394.2	118.8	149.6	349.5	81.7
399.2	118.4	149.1	349.9	81.6
404.2	118.1	148.7	350.2	81.5
409.2	117.8	148.4	350.5	81.4
414.2	118.5	149.2	349.8	81.0
419.2	118.2	148.8	350.1	80.9
424.2	117.9	148.5	350.4	80.8
429.2	117.6	148.1	350.7	80.7
434.2	118.3	148.9	350.1	80.3
439.2	118.0	148.6	350.3	80.2
444.2	117.7	148.2	350.6	80.1
449.2	117.4	147.8	350.9	79.9
454.2	118.0	148.6	350.3	79.6
459.2	117.7	148.2	350.6	79.5
464.2	117.4	147.9	350.9	79.3
469.2	117.1	147.5	351.2	79.2
474.2	117.7	148.2	350.6	78.9
479.2	117.4	147.9	350.9	78.7
484.2	117.1	147.5	351.2	81.3

Table 3.1-35 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
489.2	117.7	148.2	350.6	80.9
494.2	117.4	147.8	350.9	80.8
499.2	117.1	147.4	351.2	80.6
504.2	116.8	147.0	351.5	80.5
509.2	117.3	147.7	351.0	80.1
514.2	117.0	147.3	351.3	80.0
519.2	116.7	146.9	351.6	79.9
524.2	117.2	147.6	351.1	79.5
529.2	116.9	147.2	351.4	79.4
534.2	116.6	146.8	351.7	79.2
539.2	117.1	147.4	351.2	78.9
544.2	116.7	147.0	351.6	78.8
549.2	116.4	146.6	351.9	78.6
554.2	116.9	147.2	351.4	78.3
559.2	116.6	146.8	351.8	78.1
564.2	116.2	146.3	352.1	78.0
569.2	116.7	146.9	351.6	77.7
574.2	116.3	146.5	352.0	77.5
579.2	116.0	146.0	352.3	77.4
584.2	116.4	146.6	351.9	77.1
589.2	116.1	146.1	352.2	79.4
594.2	116.5	146.7	351.8	79.1
599.2	116.1	146.2	352.2	78.9
604.2	115.8	145.8	352.5	78.8
609.2	116.2	146.4	352.1	78.4
614.2	115.9	145.9	352.4	78.3
619.2	116.3	146.5	352.0	77.9
624.2	116.0	146.0	352.3	77.8

Table 3.1-35 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
629.2	115.6	145.6	352.7	77.6
634.2	116.0	146.1	352.3	77.3
639.2	115.7	145.6	352.6	77.1
644.2	116.1	146.1	352.3	76.8
649.2	115.7	145.7	352.6	76.6
654.2	116.0	146.1	352.3	76.3
659.2	115.7	145.6	352.7	76.1
664.2	115.3	145.1	353.1	78.4
669.2	115.6	145.6	352.7	78.1
674.2	115.2	145.0	353.1	77.9
679.2	115.5	145.4	352.8	77.5
684.2	115.8	145.8	352.5	77.2
689.2	115.4	145.3	352.9	77.0
694.2	115.7	145.6	352.7	76.7
699.2	115.2	145.1	353.1	76.6
704.2	115.5	145.4	352.9	76.2
709.2	115.0	144.8	353.3	76.1
714.2	115.2	145.1	353.1	75.7
719.2	115.4	145.3	352.9	75.4
724.2	114.9	144.7	353.4	77.6
729.2	115.1	144.9	353.2	77.3
734.2	115.2	145.1	353.1	76.9
739.2	114.7	144.5	353.6	76.8
744.2	114.8	144.6	353.5	76.4
749.2	114.9	144.7	353.4	76.1
754.2	115.0	144.8	353.3	75.8
759.2	115.1	144.9	353.3	75.5
764.2	115.1	144.9	353.2	75.2

Table 3.1-35 (Continued)
Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass and Energy Releases (BVPS-2) (Case 1L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
769.2	114.5	144.1	353.8	75.0
774.2	114.5	144.1	353.9	77.0
779.2	114.4	144.1	353.9	76.7
784.2	114.3	144.0	354.0	76.4
789.2	114.8	144.6	353.5	75.9
794.2	114.7	144.4	353.6	75.7
799.2	114.5	144.2	353.8	75.4
804.2	114.3	144.0	354.0	75.1
809.2	114.7	144.4	353.6	74.7
814.2	114.4	144.0	353.9	74.4
819.2	114.1	143.6	354.2	76.4
824.2	114.2	143.8	354.1	76.0
829.2	114.3	144.0	354.0	75.6
834.2	66.6	83.8	401.7	87.8
1089.1	66.6	83.8	401.7	87.8
1089.2	66.1	82.9	402.2	85.0
1089.2	66.1	82.9	402.2	85.0
1367.4	66.1	82.9	402.2	85.0
1367.5	57.6	66.3	410.7	13.6
2948.0	48.0	55.3	420.3	13.9
2948.1	50.5	58.1	453.9	40.0
3600.0	47.5	54.7	456.9	40.3

* Mass and Energy exiting the SG side of break

** Mass and Energy exiting the pump side of break

Table 3.1-36
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-2) (Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
21.4	.0	.0	.0	.0
21.9	.0	.0	.0	.0
22.1	.0	.0	.0	.0
22.2	.0	.0	.0	.0
22.3	.0	.0	.0	.0
22.4	.0	.0	.0	.0
22.4	.0	.0	.0	.0
22.5	57.4	67.5	.0	.0
22.6	20.7	24.4	.0	.0
22.7	18.2	21.4	.0	.0
22.9	21.6	25.4	.0	.0
23.0	31.8	37.4	.0	.0
23.1	37.0	43.6	.0	.0
23.2	43.0	50.6	.0	.0
23.3	48.3	56.8	.0	.0
23.4	53.7	63.3	.0	.0
23.5	58.3	68.6	.0	.0
23.6	61.7	72.7	.0	.0
23.7	65.1	76.7	.0	.0
23.8	66.8	78.7	.0	.0
23.8	68.4	80.6	.0	.0
23.9	71.6	84.3	.0	.0
24.0	74.7	87.9	.0	.0
24.1	77.6	91.4	.0	.0
24.2	80.5	94.8	.0	.0
24.3	83.3	98.1	.0	.0
24.4	86.0	101.3	.0	.0

Table 3.1-36 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-2) (Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
25.4	109.9	129.5	.0	.0
26.4	129.7	152.9	.0	.0
27.4	160.1	188.8	742.5	60.5
28.4	387.0	458.1	3989.5	437.1
28.5	389.5	461.0	4008.0	441.8
29.5	389.1	460.6	3995.0	446.1
30.5	382.5	452.7	3926.3	440.4
31.5	375.6	444.5	3854.3	434.0
32.5	368.7	436.3	3782.2	427.5
33.0	365.3	432.3	3746.6	424.3
33.5	362.0	428.3	3711.3	421.0
34.5	355.5	420.6	3642.1	414.7
35.5	349.3	413.2	3574.8	408.5
36.5	343.3	406.0	3509.6	402.4
37.5	337.5	399.2	3446.5	396.5
38.5	332.0	392.6	3385.3	390.8
39.2	328.2	388.1	3343.6	386.9
39.5	326.7	386.2	3326.0	385.3
40.5	321.5	380.1	3268.6	379.9
41.5	316.6	374.3	3213.0	374.7
42.5	311.8	368.6	3159.1	369.6
43.5	307.3	363.2	3106.7	364.7
44.5	302.8	357.9	3055.8	359.9
45.5	298.6	352.8	3006.4	355.3
46.3	295.3	348.9	2967.8	351.6
46.5	294.4	347.9	2958.3	350.7
47.5	290.5	343.2	2911.5	346.3
48.5	286.6	338.6	2866.0	342.0

Table 3.1-36 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases
(BVPS-2) (Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
49.5	282.8	334.1	2821.5	337.8
50.5	279.2	329.8	2778.2	333.7
51.5	275.7	325.6	2736.0	329.7
52.5	272.2	321.6	2694.7	325.7
53.5	268.9	317.6	2654.4	321.9
54.2	266.6	314.9	2626.7	319.2
54.5	265.7	313.8	2615.0	318.1
55.5	262.5	310.0	2576.4	314.4
56.5	259.4	306.4	2538.7	310.8
57.5	256.4	302.8	2501.8	307.2
58.5	253.5	299.3	2465.6	303.8
59.5	250.7	295.9	2430.1	300.3
60.5	247.9	292.6	2395.3	297.0
61.5	245.1	289.4	2361.2	293.7
62.5	242.5	286.2	2327.7	290.4
63.5	239.9	283.2	2294.9	287.2
64.5	149.2	175.9	430.0	95.7
65.5	148.9	175.5	430.6	95.5
66.5	148.6	175.2	431.2	95.4
67.5	148.3	174.8	431.8	95.2
68.5	148.0	174.5	432.5	95.1
69.5	147.7	174.1	433.1	94.9
70.5	147.4	173.8	433.8	94.8
71.5	147.1	173.4	434.4	94.6
72.5	146.8	173.1	435.1	94.5
73.5	146.5	172.7	435.7	94.3
73.6	146.5	172.7	435.8	94.3
74.5	146.2	172.4	436.4	94.2

Table 3.1-36 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-2) (Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
75.5	145.9	172.0	437.0	94.1
76.5	145.6	171.7	437.7	93.9
77.5	145.4	171.3	438.3	93.8
78.5	145.1	171.0	439.0	93.6
79.5	144.8	170.6	439.6	93.5
80.5	144.5	170.3	440.3	93.3
81.5	144.2	169.9	440.9	93.2
82.5	143.9	169.6	441.6	93.1
84.5	143.3	168.9	442.9	92.8
86.5	142.7	168.2	444.2	92.5
88.5	142.1	167.5	445.5	92.2
90.5	141.5	166.8	446.9	91.9
92.5	140.9	166.1	448.2	91.7
94.5	140.3	165.4	449.5	91.4
96.5	139.7	164.7	450.9	91.1
97.0	139.6	164.5	451.2	91.0
98.5	139.1	164.0	452.2	90.8
100.5	138.5	163.3	453.6	90.6
102.5	137.9	162.6	455.0	90.3
104.5	137.3	161.8	456.3	90.0
106.5	136.7	161.1	457.7	89.7
108.5	136.1	160.4	459.1	89.5
110.5	135.4	159.6	460.5	89.2
112.5	134.8	158.9	461.9	88.9
114.5	134.2	158.1	463.3	88.7
116.5	133.6	157.4	464.7	88.4
118.5	132.9	156.6	466.1	88.1
120.5	132.3	155.9	467.5	87.8

Table 3.1-36 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-2) (Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
122.1	131.8	155.3	468.7	87.6
122.5	131.6	155.1	468.9	87.5
124.5	131.0	154.4	470.3	87.3
126.5	130.3	153.6	471.7	87.0
128.5	129.7	152.8	473.1	86.7
130.5	129.0	152.0	474.5	86.4
132.5	128.4	151.3	475.9	86.1
134.5	127.7	150.5	477.3	85.9
136.5	127.0	149.7	478.7	85.6
138.5	126.4	148.9	480.1	85.3
140.5	125.7	148.1	481.5	85.0
142.5	125.0	147.3	482.9	84.7
144.5	124.4	146.5	484.3	84.4
146.5	123.7	145.7	485.7	84.1
148.5	123.0	144.9	487.1	83.9
149.6	122.6	144.5	487.8	83.7
150.5	122.3	144.1	488.4	83.6
152.5	121.6	143.3	489.8	83.3
154.5	120.9	142.5	491.2	83.0
156.5	120.2	141.7	492.6	82.7
158.5	119.6	140.9	494.0	82.4
160.5	118.9	140.1	495.4	82.1
162.5	118.2	139.2	496.8	81.8
164.5	117.5	138.4	498.2	81.5
166.5	116.8	137.6	499.5	81.3
168.5	116.1	136.7	500.9	81.0
170.5	115.4	135.9	502.3	80.7
172.5	114.6	135.1	503.7	80.4

Table 3.1-36 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (BVPS-2) (Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
174.5	113.9	134.2	505.1	80.1
176.5	113.2	133.4	506.4	79.8
178.5	112.5	132.6	507.8	79.5
179.8	112.1	132.0	508.7	79.3
180.5	111.8	131.7	509.2	79.2
182.5	111.3	131.1	510.2	79.2
184.5	110.8	130.5	511.1	79.1
186.5	110.3	129.9	512.0	79.1
188.5	109.8	129.3	512.9	79.1
190.5	109.3	128.7	513.8	79.0
192.5	108.8	128.1	514.7	79.0
194.5	108.3	127.6	515.6	78.9
196.5	107.8	127.0	516.5	78.9
198.5	107.3	126.4	517.3	78.8
200.5	106.8	125.8	518.2	78.8
202.5	106.3	125.3	519.1	78.7
204.5	105.8	124.7	520.0	78.7
206.5	105.3	124.1	520.8	78.6
208.5	104.9	123.5	521.7	78.6
210.5	104.4	123.0	522.6	78.5
212.5	103.9	122.4	523.4	78.4
213.4	103.7	122.2	523.8	78.4

* Mass and Energy exiting the SG side of the break

** Mass and Energy exiting the pump side of the break

Table 3.1-37
Principle Parameters During Reflood Double-Ended Pump Suction Break - Maximum Safeguards (BVPS-2) (Case 2L)

Time	Flooding		Carryover	Core Height	Downcomer	Flow	Total	Injection Accum	Spill	Enthalpy
(sec)	Temp (°F)	Rate (in/sec)	Fraction	(ft)	Height (ft)	Frac	(lbm/sec)			(Btu/lbm)
21.4	169.0	.000	.000	.00	.00	.333	.0	.0	.0	.00
22.2	166.4	22.694	.000	.67	1.40	.000	6117.9	6117.9	.0	74.50
22.4	165.1	24.146	.000	1.07	1.32	.000	6067.6	6067.6	.0	74.50
22.7	164.5	2.530	.113	1.32	1.96	.298	5952.2	5952.2	.0	74.50
22.9	164.4	2.633	.130	1.35	2.40	.310	5934.5	5934.5	.0	74.50
23.0	164.4	2.595	.163	1.37	2.69	.361	5887.9	5887.9	.0	74.50
23.2	164.4	2.604	.205	1.41	3.34	.390	5842.1	5842.1	.0	74.50
23.8	164.4	2.494	.302	1.50	4.99	.427	5710.0	5710.0	.0	74.50
24.4	164.5	2.425	.386	1.59	6.83	.442	5574.6	5574.6	.0	74.50
28.4	164.9	4.389	.635	2.01	15.61	.670	4980.9	4363.1	.0	69.35
29.5	165.0	4.190	.667	2.15	15.62	.667	4799.2	4184.3	.0	69.18
33.0	165.5	3.747	.710	2.50	15.62	.659	4434.4	3806.6	.0	68.63
39.2	167.3	3.345	.732	3.00	15.62	.644	3937.5	3290.5	.0	67.68
46.3	169.9	3.057	.740	3.50	15.62	.628	3498.4	2835.5	.0	66.64
54.2	173.4	2.826	.742	4.00	15.62	.611	3107.7	2432.0	.0	65.48
63.5	177.6	2.615	.743	4.54	15.62	.594	2731.6	2044.9	.0	64.07
64.5	178.1	2.030	.732	4.59	15.62	.483	712.1	.0	.0	33.00

Table 3.1-37 (Continued)
Principle Parameters During Reflood Double-Ended Pump Suction Break - Maximum Safeguards (BVPS-2) (Case 2L)

	Flooding						Total	Injection Accum	Spill	
Time (sec)	Temp (°F)	Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	(lbm/sec)			Enthalpy (Btu/lbm)
73.6	182.7	1.988	.733	5.00	15.62	.484	712.1	.0	.0	33.00
86.5	190.9	1.930	.735	5.56	15.62	.484	712.1	.0	.0	33.00
97.0	198.6	1.882	.736	6.00	15.62	.485	712.1	.0	.0	33.00
110.5	208.9	1.819	.739	6.55	15.62	.485	712.1	.0	.0	33.00
122.1	217.3	1.764	.740	7.00	15.62	.485	712.1	.0	.0	33.00
136.5	226.4	1.696	.743	7.54	15.62	.486	712.1	.0	.0	33.00
149.6	233.6	1.633	.744	8.00	15.62	.486	712.1	.0	.0	33.00
164.5	240.7	1.562	.746	8.51	15.62	.485	712.2	.0	.0	33.00
179.8	246.9	1.489	.747	9.00	15.62	.485	712.2	.0	.0	33.00
196.5	252.8	1.424	.749	9.51	15.62	.486	712.2	.0	.0	33.00
213.4	258.0	1.361	.751	10.00	15.62	.488	712.2	.0	.0	33.00

Table 3.1-38
Double-Ended Pump Suction Break Maximum
Safeguards Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
213.4	123.8	156.1	589.5	94.4
218.4	123.5	155.6	589.8	94.3
223.4	123.1	155.2	590.2	94.2
228.4	122.8	154.8	590.5	94.1
233.4	123.5	155.6	589.9	93.7
238.4	123.1	155.2	590.2	93.6
243.4	122.8	154.7	590.6	93.6
248.4	122.4	154.3	590.9	93.5
253.4	123.1	155.1	590.3	93.1
258.4	122.7	154.6	590.6	93.0
263.4	122.3	154.2	591.0	92.9
268.4	123.0	155.0	590.3	92.6
273.4	122.6	154.5	590.7	92.5
278.4	122.3	154.1	591.1	92.4
283.4	121.9	153.6	591.4	92.3
288.4	122.5	154.4	590.8	92.0
293.4	122.2	154.0	591.2	91.9
298.4	121.8	153.5	591.5	91.8
303.4	121.4	153.0	591.9	91.7
308.4	122.1	153.8	591.3	91.3
313.4	121.7	153.4	591.7	91.2
318.4	121.3	152.9	592.0	91.2
323.4	121.9	153.6	591.4	90.8
328.4	121.5	153.2	591.8	90.7
333.4	121.2	152.7	592.2	90.6
338.4	121.8	153.4	591.6	90.3
343.4	121.4	153.0	592.0	90.2

Table 3.1-38 (Continued)
Double-Ended Pump Suction Break Maximum
Safeguards Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
348.4	121.0	152.5	592.3	90.1
353.4	120.6	152.0	592.7	90.0
358.4	121.2	152.7	592.2	89.7
363.4	120.8	152.2	592.5	89.6
368.4	120.4	151.7	592.9	89.5
373.4	121.0	152.4	592.4	89.1
378.4	120.6	151.9	592.8	89.1
383.4	120.2	151.4	593.2	89.0
388.4	120.7	152.1	592.6	88.6
393.4	120.3	151.6	593.0	88.5
398.4	119.9	151.1	593.4	88.4
403.4	120.5	151.9	592.8	88.1
408.4	120.2	151.5	593.1	88.0
413.4	120.0	151.2	593.4	87.9
418.4	119.7	150.8	593.7	90.4
423.4	120.3	151.6	593.0	90.1
428.4	120.0	151.3	593.3	89.9
433.4	119.7	150.9	593.6	89.8
438.4	119.4	150.5	593.9	89.7
443.4	120.1	151.3	593.3	89.3
448.4	119.8	150.9	593.6	89.2
453.4	119.5	150.5	593.9	89.0
458.4	120.1	151.3	593.3	88.7
463.4	119.7	150.9	593.6	88.6
468.4	119.4	150.5	593.9	88.4
473.4	119.1	150.1	594.2	88.3
478.4	119.7	150.9	593.6	87.9
483.4	119.4	150.5	593.9	87.8

Table 3.1-38 (Continued)
Double-Ended Pump Suction Break Maximum
Safeguards Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
488.4	119.1	150.1	594.3	87.7
493.4	119.6	150.8	593.7	87.3
498.4	119.3	150.4	594.0	87.2
503.4	119.0	149.9	594.4	87.1
508.4	119.5	150.6	593.8	86.7
513.4	119.2	150.2	594.1	86.6
518.4	118.9	149.8	594.5	86.5
523.4	119.4	150.4	594.0	86.1
528.4	119.0	150.0	594.3	86.0
533.4	118.7	149.6	594.6	88.4
538.4	119.2	150.2	594.1	88.0
543.4	118.8	149.8	594.5	87.9
548.4	118.5	149.3	594.8	87.8
553.4	119.0	149.9	594.4	87.4
558.4	118.6	149.5	594.7	87.3
563.4	118.2	149.0	595.1	87.1
568.4	118.7	149.6	594.6	86.8
573.4	118.3	149.1	595.0	86.6
578.4	118.8	149.7	594.6	86.3
583.4	118.4	149.2	595.0	86.1
588.4	118.0	148.7	595.3	86.0
593.4	118.4	149.2	594.9	85.7
598.4	118.0	148.7	595.3	85.5
603.4	118.4	149.3	594.9	85.2
608.4	118.1	148.8	595.3	85.0
613.4	117.7	148.3	595.6	84.9
618.4	118.1	148.9	595.2	87.0
623.4	117.7	148.4	595.6	86.8

Table 3.1-38 (Continued)
Double-Ended Pump Suction Break Maximum
Safeguards Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
628.4	118.1	148.9	595.2	86.5
633.4	117.7	148.4	595.6	86.3
638.4	118.1	148.8	595.2	86.0
643.4	117.7	148.3	595.6	85.8
648.4	118.1	148.8	595.3	85.5
653.4	117.6	148.3	595.7	85.3
658.4	118.0	148.7	595.4	85.0
663.4	117.5	148.1	595.8	84.8
668.4	117.8	148.5	595.5	84.5
673.4	117.4	147.9	596.0	84.3
678.4	117.7	148.3	595.7	84.0
683.4	117.2	147.7	596.1	86.2
688.4	117.4	148.0	595.9	85.9
693.4	117.7	148.3	595.7	85.5
698.4	117.2	147.7	596.2	85.4
703.4	117.4	147.9	596.0	85.0
708.4	117.6	148.1	595.8	84.7
713.4	117.0	147.5	596.3	84.6
718.4	117.2	147.7	596.2	84.2
723.4	117.3	147.8	596.0	83.9
728.4	117.4	147.9	595.9	83.6
733.4	116.8	147.2	596.5	83.5
738.4	116.9	147.3	596.5	85.4
743.4	116.9	147.3	596.4	85.1
748.4	116.9	147.4	596.4	84.8
753.4	116.9	147.4	596.4	84.5
758.4	116.9	147.3	596.5	84.2
763.4	116.8	147.2	596.5	83.9

Table 3.1-38 (Continued)
Double-Ended Pump Suction Break Maximum
Safeguards Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
768.4	116.7	147.1	596.6	83.6
773.4	116.6	146.9	596.7	83.3
778.4	116.4	146.7	596.9	83.0
783.4	116.8	147.2	596.5	84.8
788.4	116.6	146.9	596.8	84.6
793.4	116.3	146.5	597.1	84.3
798.4	116.5	146.8	596.8	83.9
803.4	116.7	147.1	596.6	83.5
808.4	116.3	146.5	597.1	83.3
813.4	116.3	146.6	597.0	82.9
818.4	116.3	146.6	597.0	82.6
823.4	66.8	84.1	646.6	97.4
1079.6	66.8	84.1	646.6	97.4
1079.7	65.7	82.4	647.7	93.9
1083.4	65.6	82.4	647.7	94.4
1390.6	65.6	82.4	647.7	94.4
1390.7	55.7	64.0	657.7	21.7
2782.0	47.1	54.1	666.3	22.0
2782.1	49.5	57.0	454.9	40.1
3600.0	45.7	52.6	458.7	40.4

* Mass and Energy exiting the SG side of the break
** Mass and Energy exiting the pump side of the break

Table 3.1-39
Double-Ended Hot-Leg Break
Mass Balance (BVPS-2) (Case 3L)

		Time (Sec)		
		.00	19.20	19.20*
		Mass (Thousand lbm)		
Initial	In RCS and ACC	626.47	626.47	626.47
Added Mass	Pumped Injection	.00	.00	.00
	Total Added	.00	.00	.00
Total Available		626.47	626.47	626.47
Distribution	Reactor Coolant	416.98	49.96	49.96
	Accumulator	209.49	173.99	173.99
	Total Contents	626.47	223.96	223.96
Effluent	Break Flow	.00	402.50	402.50
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	402.50	402.50
Total Accountable		626.47	626.45	626.45

* This time is the bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

Table 3.1-40
Double-Ended Pump Suction Break Mass Balance Minimum Safeguards
(BVPS-2) (Case 1L)

		Time (Sec)						
		.00	21.40(1)	21.40(2)	214.19(3)	1089.17(4)	1367.37(5)	3600.0(6)
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	626.47	626.47	626.47	626.47	626.47	626.47	626.47
Added Mass	Pumped Injection	.00	.00	.00	85.24	495.00	625.28	1694.38
	Total Added	.00	.00	.00	85.24	495.00	625.28	1694.38
Total Available		626.47	626.47	626.47	711.71	1121.47	1251.76	2320.85
Distribution	Reactor Coolant	416.98	40.95	66.41	117.33	117.33	117.33	117.33
	Accumulator	209.49	167.73	142.28	.00	.00	.00	.00
	Total Contents	626.47	208.68	208.68	117.33	117.33	117.33	117.33
Effluent	Break Flow	.00	417.78	417.78	585.53	995.29	1125.57	2194.67
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	417.78	417.78	585.53	995.29	1125.57	2194.67
Total Accountable		626.47	626.46	626.46	702.86	1112.62	1242.91	2312.00

Notes:

(1) End of Blowdown

(2) Bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reload

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure

(6) Time at which both SGs equilibrate to 14.7 psia

Table 3.1-41
Double-Ended Pump Suction Break Mass Balance Maximum Safeguards
(BVPS-2) (Case 2L)

		Time (Sec)						
		.00	21.40[1]	21.40[2]	213.36[3]	1079.74[4]	1390.62[5]	3600.00[6]
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	626.47	626.47	626.47	626.47	626.47	626.47	626.47
Added Mass	Pumped Injection	.00	.00	.00	130.69	748.68	970.44	2375.56
	Total Added	.00	.00	.00	130.69	748.68	970.44	2375.56
Total Available		626.47	626.47	626.47	757.16	1375.15	1596.91	3002.04
Distribution	Reactor Coolant	416.98	40.95	66.41	117.53	117.53	117.53	117.53
	Accumulator	209.49	167.73	142.28	.00	.00	.00	.00
	Total Contents	626.47	208.68	208.68	117.53	117.53	117.53	117.53
Effluent	Break Flow	.00	417.78	417.78	630.78	1248.77	1470.53	2875.67
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	417.78	417.78	630.78	1248.77	1470.53	2875.67
Total Accountable		626.47	626.46	626.46	748.31	1366.30	1588.06	2993.20

Notes:
(1) End of Blowdown
(2) Bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill
(3) End of Reflood
(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure.
(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure.
(6) Time at which both SGs equilibrate to 14.7 psia.

Table 3.1-42
Double-Ended Hot-Leg Break Energy Balance (BVPS-2) (Case 3L)

		Time (Sec)		
		.00	19.20	19.20*
		Energy (Million Btu)		
Initial Energy	In RCS, Acc, SG	675.94	675.94	675.94
Added Energy	Pumped Injection	.00	.00	.00
	Decay Heat	.00	5.77	5.77
	Heat From Secondary	.00	-.35	-.35
	Total Added	.00	5.42	5.42
Total Available		675.94	681.36	681.36
Distribution	Reactor Coolant	245.25	11.99	11.99
	Accumulator	15.62	12.97	12.97
	Core Stored	22.87	9.19	9.19
	Primary Metal	115.85	108.47	108.47
	Secondary Metal	69.35	69.00	69.00
	Steam Generator	207.00	205.72	205.72
	Total Contents	675.94	417.34	417.34
Effluent	Break Flow	.00	263.52	263.52
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	263.52	263.52
Total Accountable		675.94	680.86	680.86

* This time is the bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

Table 3.1-43
Double-Ended Pump Suction Break Energy Balance Minimum Safeguards
(BVPS-2) (Case 1L)

		Time (Sec)						
		.00	21.40(1)	21.40(2)	214.19(3)	1089.17(4)	1367.37(5)	3600.00(6)
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	640.99	640.99	640.99	640.99	640.99	640.99	640.99
Added Energy	Pumped Injection	.00	.00	.00	2.81	16.33	20.63	74.05
	Decay Heat	.00	5.79	5.79	24.82	85.54	101.78	209.46
	Heat From Secondary	.00	.43	.43	.43	9.95	10.04	10.04
	Total Added	.00	6.22	6.22	28.06	111.83	132.46	293.56
Total Available		640.99	647.21	647.21	669.04	752.81	773.45	934.54
Distribution	Reactor Coolant	245.25	8.59	10.49	29.17	29.17	29.17	29.17
	Accumulator	15.62	12.51	10.61	.00	.00	.00	.00
	Core Stored	22.87	12.50	12.50	3.91	3.17	3.12	2.71
	Primary Metal	115.85	109.61	109.61	88.92	50.61	45.22	40.05
	Secondary Metal	34.40	34.82	34.82	31.86	19.12	16.34	14.61
	Steam Generator	207.00	210.21	210.21	188.90	116.55	100.61	90.81
	Total Contents	640.99	388.24	388.24	342.76	218.61	194.45	177.35
Effluent	Break Flow	.00	258.48	258.48	318.23	526.15	549.53	730.23
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	258.48	258.48	318.23	526.15	549.53	730.23
Total Accountable		640.99	646.72	646.72	660.99	744.76	743.98	907.58

Notes:

(1) End of Blowdown.

(2) Bottom of core recovery time. This time is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reload.

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure

(6) Time at which both SGs equilibrate to 14.7 psia.

Table 3.1-44
Double-Ended Pump Suction Break Energy Balance - Maximum Safeguards (BVPS-2) (Case 2L)

		Time (sec)						
		.00	$21.40^{(1)}$	$21.40^{(2)}$	$213.36^{(3)}$	$1079.74^{(4)}$	$1390.62^{(5)}$	$3600.00^{(6)}$
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	640.99	640.99	640.99	640.99	640.99	640.99	640.99
Added Energy	Pumped Injection	.00	.00	.00	4.31	24.71	32.02	101.15
	Decay Heat	.00	5.79	5.79	24.75	84.97	103.09	209.42
	Heat From Secondary	.00	.43	.43	.43	9.86	9.93	9.93
	Total Added	.00	6.22	6.22	29.49	119.54	145.04	320.50
Total Available		640.99	647.21	647.21	670.47	760.52	786.02	961.49
Distribution	Reactor Coolant	245.25	8.59	10.49	29.17	29.17	29.17	29.17
	Accumulator	15.62	12.51	10.61	.00	.00	.00	.00
	Core Stored	22.87	12.50	12.50	3.91	3.10	3.05	2.71
	Primary Metal	115.85	109.61	109.61	88.61	49.94	43.94	39.96
	Secondary Metal	34.40	34.82	34.82	31.84	18.94	15.82	14.56
	Steam Generator	207.00	210.21	210.21	188.76	115.44	97.55	90.42
	Total Contents	640.99	388.24	388.24	342.31	216.59	189.53	176.82
Effluent	Break Flow	.00	258.48	258.48	320.12	535.87	566.71	757.09
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	258.48	258.48	320.12	535.87	566.71	757.09
Total Accountable		640.99	646.72	646.72	662.42	752.47	756.25	933.91

Notes:
(1) End of Blowdown.
(2) Bottom of core recovery time. This time is identical to the end of blowdown time due to the assumption of instantaneous refill.
(3) End of Reflood.
(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure.
(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure.
(6) Time at which both SGs equilibrate to 14.7 psia.

Table 3.1-45
Double-Ended Pump Suction Break (SW Failure) - Maximum Safeguards Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
21.4	.0	.0	.0	.0
21.9	.0	.0	.0	.0
22.1	.0	.0	.0	.0
22.2	.0	.0	.0	.0
22.3	.0	.0	.0	.0
22.4	.0	.0	.0	.0
22.4	.0	.0	.0	.0
22.5	57.4	67.5	.0	.0
22.6	20.7	24.4	.0	.0
22.7	18.2	21.4	.0	.0
22.9	21.6	25.4	.0	.0
23.0	31.8	37.4	.0	.0
23.1	37.0	43.6	.0	.0
23.2	43.0	50.6	.0	.0
23.3	48.3	56.8	.0	.0
23.4	53.7	63.3	.0	.0
23.5	58.3	68.6	.0	.0
23.6	61.7	72.7	.0	.0
23.7	65.1	76.7	.0	.0
23.8	66.8	78.7	.0	.0
23.8	68.4	80.6	.0	.0
23.9	71.6	84.3	.0	.0
24.0	74.7	87.9	.0	.0
24.1	77.6	91.4	.0	.0
24.2	80.5	94.8	.0	.0
24.3	83.3	98.1	.0	.0
24.4	86.0	101.3	.0	.0
25.4	109.9	129.5	.0	.0

Table 3.1-45 (Continued)
Double-Ended Pump Suction Break (SW Failure) - Maximum Safeguards Reflood
Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
26.4	129.7	152.9	.0	.0
27.4	160.1	188.8	742.5	60.5
28.4	387.0	458.1	3989.5	437.1
28.5	389.5	461.0	4008.0	441.8
29.5	389.1	460.6	3995.0	446.1
30.5	382.5	452.7	3926.3	440.4
31.5	375.6	444.5	3854.3	434.0
32.5	368.7	436.3	3782.2	427.5
33.0	365.3	432.3	3746.6	424.3
33.5	362.0	428.3	3711.3	421.0
34.5	355.5	420.6	3642.1	414.7
35.5	349.3	413.2	3574.8	408.5
36.5	343.3	406.0	3509.6	402.4
37.5	337.5	399.2	3446.5	396.5
38.5	332.0	392.6	3385.3	390.8
39.2	328.2	388.1	3343.6	386.9
39.5	326.7	386.2	3326.0	385.3
40.5	321.5	380.1	3268.6	379.9
41.5	316.6	374.3	3213.0	374.7
42.5	311.8	368.6	3159.1	369.6
43.5	307.3	363.2	3106.7	364.7
44.5	302.8	357.9	3055.8	359.9
45.5	298.6	352.8	3006.4	355.3
46.3	295.3	348.9	2967.8	351.6
46.5	294.4	347.9	2958.3	350.7
47.5	290.5	343.2	2911.5	346.3
48.5	286.6	338.6	2866.0	342.0
49.5	282.8	334.1	2821.5	337.8

Table 3.1-45 (Continued)
Double-Ended Pump Suction Break (SW Failure) - Maximum Safeguards Reflood
Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
50.5	279.2	329.8	2778.2	333.7
51.5	275.7	325.6	2736.0	329.7
52.5	272.2	321.6	2694.7	325.7
53.5	268.9	317.6	2654.4	321.9
54.2	266.6	314.9	2626.7	319.2
54.5	265.7	313.8	2615.0	318.1
55.5	262.5	310.0	2576.4	314.4
56.5	259.4	306.4	2538.7	310.8
57.5	256.4	302.8	2501.8	307.2
58.5	253.5	299.3	2465.6	303.8
59.5	250.7	295.9	2430.1	300.3
60.5	247.9	292.6	2395.3	297.0
61.5	245.1	289.4	2361.2	293.7
62.5	242.5	286.2	2327.7	290.4
63.5	239.9	283.2	2294.9	287.2
64.5	149.2	175.9	430.0	95.7
65.5	148.9	175.5	430.6	95.5
66.5	148.6	175.2	431.2	95.4
67.5	148.3	174.8	431.8	95.2
68.5	148.0	174.5	432.5	95.1
69.5	147.7	174.1	433.1	94.9
70.5	147.4	173.8	433.8	94.8
71.5	147.1	173.4	434.4	94.6
72.5	146.8	173.1	435.1	94.5
73.5	146.5	172.7	435.7	94.3
73.6	146.5	172.7	435.8	94.3
74.5	146.2	172.4	436.4	94.2
75.5	145.9	172.0	437.0	94.1

Table 3.1-45 (Continued)
Double-Ended Pump Suction Break (SW Failure) - Maximum Safeguards Reflood
Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
76.5	145.6	171.7	437.7	93.9
77.5	145.4	171.3	438.3	93.8
78.5	145.1	171.0	439.0	93.6
79.5	144.8	170.6	439.6	93.5
80.5	144.5	170.3	440.3	93.3
81.5	144.2	169.9	440.9	93.2
82.5	143.9	169.6	441.6	93.1
84.5	143.3	168.9	442.9	92.8
86.5	142.7	168.2	444.2	92.5
88.5	142.1	167.5	445.5	92.2
90.5	141.5	166.8	446.9	91.9
92.5	140.9	166.1	448.2	91.7
94.5	140.3	165.4	449.5	91.4
96.5	139.7	164.7	450.9	91.1
97.0	139.6	164.5	451.2	91.0
98.5	139.1	164.0	452.2	90.8
100.5	138.5	163.3	453.6	90.6
102.5	137.9	162.6	455.0	90.3
104.5	137.3	161.8	456.3	90.0
106.5	136.7	161.1	457.7	89.7
108.5	136.1	160.4	459.1	89.5
110.5	135.4	159.6	460.5	89.2
112.5	134.8	158.9	461.9	88.9
114.5	134.2	158.1	463.3	88.7
116.5	133.6	157.4	464.7	88.4
118.5	132.9	156.6	466.1	88.1
120.5	132.3	155.9	467.5	87.8
122.1	131.8	155.3	468.7	87.6

Table 3.1-45 (Continued)
Double-Ended Pump Suction Break (SW Failure) - Maximum Safeguards Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
122.5	131.6	155.1	468.9	87.5
124.5	131.0	154.4	470.3	87.3
126.5	130.3	153.6	471.7	87.0
128.5	129.7	152.8	473.1	86.7
130.5	129.0	152.0	474.5	86.4
132.5	128.4	151.3	475.9	86.1
134.5	127.7	150.5	477.3	85.9
136.5	127.0	149.7	478.7	85.6
138.5	126.4	148.9	480.1	85.3
140.5	125.7	148.1	481.5	85.0
142.5	125.0	147.3	482.9	84.7
144.5	124.4	146.5	484.3	84.4
146.5	123.7	145.7	485.7	84.1
148.5	123.0	144.9	487.1	83.9
149.6	122.6	144.5	487.8	83.7
150.5	122.3	144.1	488.4	83.6
152.5	121.6	143.3	489.8	83.3
154.5	120.9	142.5	491.2	83.0
156.5	120.2	141.7	492.6	82.7
158.5	119.6	140.9	494.0	82.4
160.5	118.9	140.1	495.4	82.1
162.5	118.2	139.2	496.8	81.8
164.5	117.5	138.4	498.2	81.5
166.5	116.8	137.6	499.5	81.3
168.5	116.1	136.7	500.9	81.0
170.5	115.4	135.9	502.3	80.7
172.5	114.6	135.1	503.7	80.4
174.5	113.9	134.2	505.1	80.1

Table 3.1-45 (Continued)
Double-Ended Pump Suction Break (SW Failure) - Maximum Safeguards Reflood
Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
176.5	113.2	133.4	506.4	79.8
178.5	112.5	132.6	507.8	79.5
179.8	112.1	132.0	508.7	79.3
180.5	111.8	131.7	509.2	79.2
182.5	111.3	131.1	510.2	79.2
184.5	110.8	130.5	511.1	79.1
186.5	110.3	129.9	512.0	79.1
188.5	109.8	129.3	512.9	79.1
190.5	109.3	128.7	513.8	79.0
192.5	108.8	128.1	514.7	79.0
194.5	108.3	127.6	515.6	78.9
196.5	107.8	127.0	516.5	78.9
198.5	107.3	126.4	517.3	78.8
200.5	106.8	125.8	518.2	78.8
202.5	106.3	125.3	519.1	78.7
204.5	105.8	124.7	520.0	78.7
206.5	105.3	124.1	520.8	78.6
208.5	104.9	123.5	521.7	78.6
210.5	104.4	123.0	522.6	78.5
212.5	103.9	122.4	523.4	78.4
213.4	103.7	122.2	523.8	78.4

* Mass and Energy exiting the SG side of break

** Mass and Energy exiting the pump side of break

Table 3.1-46
Double-Ended Pump Suction (SW Failure) - Maximum Safeguards Principle Parameters During Reflood
(BVPS-2) (Case 2L1)

Time (sec)	Flooding		Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total	Injection Accum	Spill	Enthalpy (Btu/lbm)
	Temp (°F)	Rate (in/sec)					(lbm/sec)			
21.4	169.0	.000	.000	.00	.00	.333	.0	.0	.0	.00
22.2	166.4	22.694	.000	.67	1.40	.000	6117.9	6117.9	.0	74.50
22.4	165.1	24.146	.000	1.07	1.32	.000	6067.6	6067.6	.0	74.50
22.7	164.5	2.530	.113	1.32	1.96	.298	5952.2	5952.2	.0	74.50
22.9	164.4	2.633	.130	1.35	2.40	.310	5934.5	5934.5	.0	74.50
23.0	164.4	2.595	.163	1.37	2.69	.361	5887.9	5887.9	.0	74.50
23.2	164.4	2.604	.205	1.41	3.34	.390	5842.1	5842.1	.0	74.50
23.8	164.4	2.494	.302	1.50	4.99	.427	5710.0	5710.0	.0	74.50
24.4	164.5	2.425	.386	1.59	6.83	.442	5574.6	5574.6	.0	74.50
28.4	164.9	4.389	.635	2.01	15.61	.670	4980.9	4363.1	.0	69.35
29.5	165.0	4.190	.667	2.15	15.62	.667	4799.2	4184.3	.0	69.18
33.0	165.5	3.747	.710	2.50	15.62	.659	4434.4	3806.6	.0	68.63
39.2	167.3	3.345	.732	3.00	15.62	.644	3937.5	3290.5	.0	67.68
46.3	169.9	3.057	.740	3.50	15.62	.628	3498.4	2835.5	.0	66.64
54.2	173.4	2.826	.742	4.00	15.62	.611	3107.7	2432.0	.0	65.48
63.5	177.6	2.615	.743	4.54	15.62	.594	2731.6	2044.9	.0	64.07
64.5	178.1	2.030	.732	4.59	15.62	.483	712.1	.0	.0	33.00

Table 3.1-46 (Continued)
Double-Ended Pump Suction (SW Failure) - Maximum Safeguards Principle Parameters During Reflood (BVPS-2) (Case 2L1)

Time (sec)	Flooding		Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total	Injection Accum	Spill	Enthalpy (Btu/lbm)
	Temp (°F)	Rate (in/sec)					(lbm/sec)			
73.6	182.7	1.988	.733	5.00	15.62	.484	712.1	.0	.0	33.00
86.5	190.9	1.930	.735	5.56	15.62	.484	712.1	.0	.0	33.00
97.0	198.6	1.882	.736	6.00	15.62	.485	712.1	.0	.0	33.00
110.5	208.9	1.819	.739	6.55	15.62	.485	712.1	.0	.0	33.00
122.1	217.3	1.764	.740	7.00	15.62	.485	712.1	.0	.0	33.00
136.5	226.4	1.696	.743	7.54	15.62	.486	712.1	.0	.0	33.00
149.6	233.6	1.633	.744	8.00	15.62	.486	712.1	.0	.0	33.00
164.5	240.7	1.562	.746	8.51	15.62	.485	712.2	.0	.0	33.00
179.8	246.9	1.489	.747	9.00	15.62	.485	712.2	.0	.0	33.00
213.4	258.0	1.361	.751	10.00	15.62	.488	712.2	.0	.0	33.00

Table 3.1-47
Double-Ended Pump Suction (SW Failure) Maximum Safeguards
Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* (Thousand Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** (Thousand Btu/sec)
213.4	123.8	156.1	589.5	94.4
218.4	123.5	155.6	589.8	94.3
223.4	123.1	155.2	590.2	94.2
228.4	122.8	154.8	590.5	94.1
233.4	123.5	155.6	589.9	93.7
238.4	123.1	155.2	590.2	93.6
243.4	122.8	154.7	590.6	93.6
248.4	122.4	154.3	590.9	93.5
253.4	123.1	155.1	590.3	93.1
258.4	122.7	154.6	590.6	93.0
263.4	122.3	154.2	591.0	92.9
268.4	123.0	155.0	590.3	92.6
273.4	122.6	154.5	590.7	92.5
278.4	122.3	154.1	591.1	92.4
283.4	121.9	153.6	591.4	92.3
288.4	122.5	154.4	590.8	92.0
293.4	122.2	154.0	591.2	91.9
298.4	121.8	153.5	591.5	91.8
303.4	121.4	153.0	591.9	91.7
308.4	122.1	153.8	591.3	91.3
313.4	121.7	153.4	591.7	91.2
318.4	121.3	152.9	592.0	91.2
323.4	121.9	153.6	591.4	90.8
328.4	121.5	153.2	591.8	90.7
333.4	121.2	152.7	592.2	90.6
338.4	121.8	153.4	591.6	90.3
343.4	121.4	153.0	592.0	90.2

Table 3.1-47 (Continued)
Double-Ended Pump Suction (SW Failure) Maximum Safeguards
Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
348.4	121.0	152.5	592.3	90.1
353.4	120.6	152.0	592.7	90.0
358.4	121.2	152.7	592.2	89.7
363.4	120.8	152.2	592.5	89.6
368.4	120.4	151.7	592.9	89.5
373.4	121.0	152.4	592.4	89.1
378.4	120.6	151.9	592.8	89.1
383.4	120.2	151.4	593.2	89.0
388.4	120.7	152.1	592.6	88.6
393.4	120.3	151.6	593.0	88.5
398.4	119.9	151.1	593.4	88.4
403.4	120.5	151.9	592.8	88.1
408.4	120.2	151.5	593.1	88.0
413.4	120.0	151.2	593.4	87.9
418.4	119.7	150.8	593.7	90.4
423.4	120.3	151.6	593.0	90.1
428.4	120.0	151.3	593.3	89.9
433.4	119.7	150.9	593.6	89.8
438.4	119.4	150.5	593.9	89.7
443.4	120.1	151.3	593.3	89.3
448.4	119.8	150.9	593.6	89.2
453.4	119.5	150.5	593.9	89.0
458.4	120.1	151.3	593.3	88.7
463.4	119.7	150.9	593.6	88.6
468.4	119.4	150.5	593.9	88.4
473.4	119.1	150.1	594.2	88.3
478.4	119.7	150.9	593.6	87.9
483.4	119.4	150.5	593.9	87.8

Table 3.1-47 (Continued)
Double-Ended Pump Suction (SW Failure) Maximum Safeguards
Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
488.4	119.1	150.1	594.3	87.7
493.4	119.6	150.8	593.7	87.3
498.4	119.3	150.4	594.0	87.2
503.4	119.0	149.9	594.4	87.1
508.4	119.5	150.6	593.8	86.7
513.4	119.2	150.2	594.1	86.6
518.4	118.9	149.8	594.5	86.5
523.4	119.4	150.4	594.0	86.1
528.4	119.0	150.0	594.3	86.0
533.4	118.7	149.6	594.6	88.4
538.4	119.2	150.2	594.1	88.0
543.4	118.8	149.8	594.5	87.9
548.4	118.5	149.3	594.8	87.8
553.4	119.0	149.9	594.4	87.4
558.4	118.6	149.5	594.7	87.3
563.4	118.2	149.0	595.1	87.1
568.4	118.7	149.6	594.6	86.8
573.4	118.3	149.1	595.0	86.6
578.4	118.8	149.7	594.6	86.3
583.4	118.4	149.2	595.0	86.1
588.4	118.0	148.7	595.3	86.0
593.4	118.4	149.2	594.9	85.7
598.4	118.0	148.7	595.3	85.5
603.4	118.4	149.3	594.9	85.2
608.4	118.1	148.8	595.3	85.0
613.4	117.7	148.3	595.6	84.9
618.4	118.1	148.9	595.2	87.0
623.4	117.7	148.4	595.6	86.8

Table 3.1-47 (Continued)
Double-Ended Pump Suction (SW Failure) Maximum Safeguards
Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
628.4	118.1	148.9	595.2	86.5
633.4	117.7	148.4	595.6	86.3
638.4	118.1	148.8	595.2	86.0
643.4	117.7	148.3	595.6	85.8
648.4	118.1	148.8	595.3	85.5
653.4	117.6	148.3	595.7	85.3
658.4	118.0	148.7	595.4	85.0
663.4	117.5	148.1	595.8	84.8
668.4	117.8	148.5	595.5	84.5
673.4	117.4	147.9	596.0	84.3
678.4	117.7	148.3	595.7	84.0
683.4	117.2	147.7	596.1	86.2
688.4	117.4	148.0	595.9	85.9
693.4	117.7	148.3	595.7	85.5
698.4	117.2	147.7	596.2	85.4
703.4	117.4	147.9	596.0	85.0
708.4	117.6	148.1	595.8	84.7
713.4	117.0	147.5	596.3	84.6
718.4	117.2	147.7	596.2	84.2
723.4	117.3	147.8	596.0	83.9
728.4	117.4	147.9	595.9	83.6
733.4	116.8	147.2	596.5	83.5
738.4	116.9	147.3	596.5	85.4
743.4	116.9	147.3	596.4	85.1
748.4	116.9	147.4	596.4	84.8
753.4	116.9	147.4	596.4	84.5
758.4	116.9	147.3	596.5	84.2
763.4	116.8	147.2	596.5	83.9

Table 3.1-47 (Continued)
Double-Ended Pump Suction (SW Failure) Maximum Safeguards
Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
768.4	116.7	147.1	596.6	83.6
773.4	116.6	146.9	596.7	83.3
778.4	116.4	146.7	596.9	83.0
783.4	116.8	147.2	596.5	84.8
788.4	116.6	146.9	596.8	84.6
793.4	116.3	146.5	597.1	84.3
798.4	116.5	146.8	596.8	83.9
803.4	116.7	147.1	596.6	83.5
808.4	116.3	146.5	597.1	83.3
813.4	116.3	146.6	597.0	82.9
818.4	116.3	146.6	597.0	82.6
823.4	66.8	84.1	646.6	97.4
1075.0	66.8	84.1	646.6	97.4
1075.1	64.8	81.5	648.5	93.4
1078.4	64.8	81.4	648.6	93.9
1403.5	64.8	81.4	648.6	93.9
1403.6	54.3	62.5	659.0	21.7
2128.0	48.7	56.0	664.6	21.9
2128.1	52.7	60.7	781.3	92.2
3600.0	45.7	52.6	788.3	93.0

* Mass and Energy exiting the SG side of break
** Mass and Energy exiting the pump side of break

Table 3.1-48
Double-Ended Pump Suction (SW Failure) Mass Balance
Maximum Safeguards (BVPS-2) (Case 2L1)

		Time (Sec)						
		.00	21.40(1)	21.40(2)	213.36(3)	1075.07(4)	1403.54(5)	3600.00(6)
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	626.47	626.47	626.47	626.47	626.47	626.47	626.47
Added Mass	Pumped Injection	.00	.00	.00	130.69	745.35	979.66	2724.09
	Total Added	.00	.00	.00	130.69	745.35	979.66	2724.09
Total Available		626.47	626.47	626.47	757.16	1371.83	1606.13	3350.56
Distribution	Reactor Coolant	416.98	40.95	66.41	117.53	117.53	117.53	117.53
	Accumulator	209.49	167.73	142.28	.00	.00	.00	.00
	Total Contents	626.47	208.68	208.68	117.53	117.53	117.53	117.53
Effluent	Break Flow	.00	417.78	417.78	630.78	1245.45	1479.75	3224.18
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	417.78	417.78	630.78	1245.45	1479.75	3224.18
Total Accountable		626.47	626.46	626.46	748.31	1362.97	1597.28	3341.71

Notes:

(1) End of Blowdown

(2) Bottom of core recovery time. This time is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure.

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure.

(6) Time at which both SGs equilibrate to 14.7 psia.

Table 3.1-49
Double-Ended Pump Suction (SW Failure) Energy Balance
Maximum Safeguards (BVPS-2) (Case 2L1)

		Time (Sec)						
		.00	21.40(1)	21.40(2)	213.36(3)	1075.07(4)	1403.54(5)	3600.00(6)
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	640.99	640.99	640.99	640.99	640.99	640.99	640.99
Added Energy	Pumped Injection	.00	.00	.00	4.31	24.60	32.33	194.25
	Decay Heat	.00	5.79	5.79	24.75	84.69	103.81	209.42
	Heat From Secondary	.00	.43	.43	.43	9.81	9.93	9.93
	Total Added	.00	6.22	6.22	29.49	119.09	146.07	413.59
Total Available		640.99	647.21	647.21	670.47	760.08	787.05	1054.58
Distribution	Reactor Coolant	245.25	8.59	10.49	29.17	29.17	29.17	29.17
	Accumulator	15.62	12.51	10.61	.00	.00	.00	.00
	Core Stored	22.87	12.50	12.50	3.91	2.98	2.95	2.71
	Primary Metal	115.85	109.61	109.61	88.61	49.14	42.92	39.96
	Secondary Metal	34.40	34.82	34.82	31.84	18.80	15.53	14.56
	Steam Generator	207.00	210.21	210.21	188.76	114.65	95.92	90.42
	Total Contents	640.99	388.24	388.24	342.31	214.75	186.49	176.82
Effluent	Break Flow	.00	258.48	258.48	320.12	537.28	568.50	846.95
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	258.48	258.48	320.12	537.28	568.50	846.95
Total Accountable		640.99	646.72	646.72	662.42	752.03	754.99	1023.77

Notes:

(1) End of Blowdown.

(2) Bottom of core recovery time. This time is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reflood.

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure.

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure.

(6) Time at which both SGs equilibrate to 14.7 psia.

3.2 MSLB MASS AND ENERGY RELEASES

3.2.1 Introduction

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment and elevated containment temperatures and pressures. The magnitude of the releases following a steamline rupture is dependent upon the plant initial operating conditions and the size of the rupture as well as the configuration of the plant steam system and the containment design. These variations make it difficult to determine the absolute worst cases for either containment pressure or temperature evaluation following a steamline break. The analysis considers a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break size in determining the main steamline break (MSLB) mass and energy releases for use in containment analysis.

3.2.2 Input Parameters and Assumptions

The analysis inputs, assumptions, and methods pertaining to the main steamline break mass and energy releases inside containment are presented in this section.

To determine the effects of plant power level and break area on the mass and energy releases from a ruptured steamline, spectra of both variables have been evaluated. At nominal full NSSS power levels (see Table 3.2-1) of 100.6 percent, 70 percent, 30 percent and 0 percent of nominal full-load power, three break sizes have been defined. These break areas are defined as the following.

1. A full double-ended rupture (DER) upstream of the inline flow restrictor in one steamline. Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break displace from each other. This large DER represents the largest break of the main steamline producing the highest mass flowrate from the faulted-loop steam generator. This break location is applicable only to BVPS-1 with the Westinghouse–design Model 51 steam generators. The BVPS-1 replacement steam generators and the BVPS-2 steam generators have flow restrictors integral with the outlet nozzle.

2. A full DER downstream of the flow restrictor in one steamline. This DER represents the break of the main steamline in which the mass flowrate from the faulted-loop steam generator is limited by the cross-sectional area of the flow restrictor.

3. A small split rupture that will neither generate a steamline isolation signal from the Westinghouse Solid-State Protection System (SSPS) nor result in water entrainment in the break effluent. Reactor protection and safety injection actuation functions are obtained from containment pressure signals.

The various cases included in the analyses have been chosen based on the selection of similar steamline ruptures included in the analyses presented in the BVPS-2 Updated Final Safety Analysis Report (UFSAR), subsection 6.2.1.4. The cases, listed in subsection 3.2.3 of this licensing report, have been analyzed assuming operation with the Westinghouse-design Model 51 steam generators (BVPS-1) and the Westinghouse-design Model 51M steam generators (BVPS-2), and include a revised set of the cases previously analyzed for Beaver Valley Power Station. Also, although the results are not specifically listed in this section, BVPS-1 was analyzed for the Model 54F replacement steam generators. The containment analysis described in Chapter 4 addresses both steam generator types for BVPS-1. All cases have been analyzed at the future uprated power condition (see Table 3.2-1). Other assumptions regarding important plant conditions and features are discussed in the following paragraphs.

3.2.2.1 Initial Power Level

Steamline breaks can be postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since steam generator water mass decreases with increasing power level, breaks occurring at lower power levels will generally result in a greater total mass release to the containment. However, because of increased stored energy in the primary side of the plant, increased heat transfer in the steam generators, and additional energy generation in the fuel, the energy release to the containment from breaks postulated to occur during full-power, or near full-power, operation may be greater than for breaks occurring with the plant in a low-power, or hot-shutdown, condition. Additionally, pressure in the steam generators changes with increasing power and has a significant influence on the rate of blowdown.

Because of the opposing effects on mass versus energy release for the MSLB due to a change in initial power level, a single power level cannot be specified as the worst case for either the containment pressure cases or the containment temperature cases. Therefore, representative power levels including 100.6 percent, 70 percent, 30 percent and 0 percent of nominal full NSSS power conditions (see Table 3.2-1) have been investigated for BVPS-1 and 2, based on the information in Reference 1. Reference 1 has been reviewed and approved by the Nuclear Regulatory Commission (NRC) for use in MSLB analysis inside containment. Additional discussion is provided in subsection 3.2.3 of this report.

In general, the plant initial conditions are assumed to be at the nominal value corresponding to the initial power for that case, with appropriate uncertainties included. Tables 3.2-1 and 3.2-2 identify the values assumed for Reactor Coolant System (RCS) pressure, RCS vessel average temperature, RCS flow, pressurizer water volume, steam generator water level, steam generator pressure, and feedwater enthalpy corresponding to each power level analyzed. Steamline break mass and energy releases assuming an RCS average temperature at the high end of the T_{avg} window are conservative with respect to similar releases at the low end of the T_{avg} window. At the high end, there is more mass and energy available for release into containment. The thermal design flowrate has been used for the RCS flow input consistent with the assumptions documented in Reference 1. The thermal design flowrate is also consistent with other MSLB analysis assumptions related to nonstatistical treatment of uncertainties, as well as RCS thermal-hydraulic inputs related to pressure drops and rod drop time.

Uncertainties on the initial conditions assumed in the analysis for the Beaver Valley Power Station EPU analysis program have been applied only to the RCS average temperature (8.5°F), the steam generator mass (7 percent narrow-range span) and the power fraction (0.6 percent) at full power. Nominal values are adequate for the initial conditions associated with pressurizer pressure and pressurizer water level. Uncertainty conditions are only applied to those parameters that could increase the amount of mass or energy discharged into containment.

3.2.2.2 Single-Failure Assumptions

In a manner consistent with the standard approach for licensing-basis analyses, various single failures have been identified and used in the spectrum of MSLB cases analyzed. Most cases analyzed considered only one single failure. One of these failures is considered as part of the containment response analysis as discussed in Section 4.1. The postulated single failures (discussed also in Reference 1) that increase the MSLB mass and energy releases to containment are discussed below.

1. Failure of the Main Steam Check Valve (MSCV) in the Faulted Loop (BVPS-1 only)

   Reverse steam flow in each of the three steamlines is prevented via the MSCV. Each valve closes automatically when a force is exerted on the downstream side of the valve, to prevent steam flow backwards from the main steam header and the other two steam generators. The main steamline rupture upstream of this valve, as postulated for the inside-containment analysis, creates a situation in which the steam generator on the faulted loop cannot be isolated, even when the MSCV is closed. The break location allows a continued blowdown from the faulted-loop steam generator until it is empty and all sources of feedwater and auxiliary feedwater addition are terminated. If the faulted-loop MSCV fails to close, blowdown from more than one steam generator can occur, but is terminated by the closure of the main steam isolation valve (MSIV) for each intact-loop steam generator. However, there is no failure of a single MSCV that could cause continued blowdown from multiple steam generators.

   In addition to the continued blowdown from the faulted-loop steam generator after MSCV closure, the steam in the unisolable section of the steamline needs to be considered. An MSCV failure can impact the mass and energy releases, since a failed MSCV will result in a larger unisolable steamline volume. The analytical method of addressing the steamline piping blowdown and the effect of an MSCV failure is dependent on break type, as discussed in subsection 3.2.3.

2. Failure of the Main Steam Isolation Valve (MSIV) in the Faulted Loop (BVPS-2 only)

   For BVPS-2, which does not have MSCVs, the main steamline isolation function is accomplished via the MSIV in each of the three steamlines. Each valve closes on an isolation signal to terminate steam flow from the associated steam generator. The main steamline rupture upstream of this valve, as postulated for the inside-containment analysis, creates a situation in which the steam generator on the faulted loop cannot be isolated, even when the MSIV successfully closes. The break location allows a continued blowdown from the faulted-loop steam generator until it is empty and all sources of feedwater and auxiliary feedwater addition are terminated. If the faulted-loop MSIV fails to close, blowdown from more than one steam generator is terminated by the closure of the corresponding MSIV for each intact-loop steam generator. Therefore, there is no failure of a single MSIV that could cause continued blowdown from multiple steam generators.

   In addition to the continued blowdown from the faulted-loop steam generator after MSIV closure, the steam in the unisolable section of the steamline needs to be considered. An MSIV failure can impact the mass and energy releases, since a failed MSIV will result in a larger unisolable steamline volume. The analytical method of addressing the steamline piping blowdown and the effect of an MSIV failure is dependent on break type, as discussed in subsection 3.2.3.

3. Failure of the Main Feedwater Isolation Valve (FIV) in the Faulted Loop (BVPS-1 only)

   If the FIV in the feedwater line to the faulted steam generator is assumed to fail in the open position, backup isolation is provided via the main feedwater regulator valve (FRV) closure. The additional inventory between the FIV and the FRV in the faulted loop would be available to be released to containment. The FIV for each main feedwater line is a new valve, and will be located to minimize the unisolable volume between the FIV and the FRV in the faulted loop. The failure of the FIV also increases the delay time for feedwater isolation for the 0% power steamline rupture cases, since the bypass FRV used at this power level has a longer closing time. Because of the relatively small effect of the FIV failure (when the FIV and FRV are near each other), this single failure is not always separately analyzed and may conservatively be combined with other single-failure cases.

4. Failure of the Main Feedwater Isolation Valve (FIV) in the Faulted Loop (BVPS-2 only)

   If the FIV in the feedwater line to the faulted steam generator is assumed to fail in the open position, backup isolation is provided via the main feedwater flow control valve (FCV) closure. The additional inventory between the FIV and the FCV in the faulted loop would be available to be released to containment.

3.2.2.3 Main Feedwater System

The rapid depressurization that occurs following a steamline rupture typically results in large amounts of water being added to the steam generators through the main feedwater system. Rapid-closing FRVs, FIVs, or FCVs in the main feedwater lines limit this effect. The feedwater addition that occurs prior to closing of the FRVs, FIVs, or FCVs influences the steam generator blowdown in several ways. First, because the water entering the steam generator is subcooled, it lowers the steam pressure thereby reducing the flowrate out of the break. As the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the isolation valves will flash into the steam generators providing additional secondary fluid which may exit out of the rupture. Secondly, the increased flow causes an increase in the total heat transfer from the primary to secondary systems resulting in greater integrated energy being released out of the break.

Following the initiation of the MSLB, main feedwater flow is conservatively modeled, as shown in Table 3.2-3. The initial increase in feedwater flow (until fully isolated) is in response to the feedwater control valve opening up in response to the steam flow/feedwater flow mismatch, or the decreasing steam generator water level as well as due to a lower backpressure on the feedwater pump as a result of the depressurizing steam generator. This maximizes the total mass addition prior to feedwater isolation. The feedwater isolation response time, following the safety injection signal, is assumed to be a total of 10 seconds (for BVPS-1) or 7 seconds (for BVPS-2), accounting for delays associated with signal processing plus FIV stroke time. For the circumstance in which the FIV in the faulted loop fails to close for BVPS-1, there is no effect on the feedwater isolation response time for the at-power cases since the total delay for the FRV closure is also assumed to be 10 seconds. However, at zero-power initial conditions, when the assumption of an FIV failure is made, the total delay for the FRV bypass closure is assumed to be 30 seconds. For the circumstance in which the FIV in the faulted loop fails to close for BVPS-2, there is no effect on the feedwater isolation response time since the total delay for the FCV closure is also 7 seconds, regardless of the power level.

Following feedwater isolation, as the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the isolation, regulator, or control valve may flash to steam if the feedwater temperature exceeds the saturation temperature. This unisolable feedwater line volume is an additional source of fluid that can increase the mass discharged out of the break. The unisolable volume in the feedwater lines is maximized for the faulted loop. The feedwater line piping volume available for steam flashing in this analysis is shown in Table 3.2-3.

Steamline break mass and energy releases assuming a main feedwater temperature at the high end of the feedwater temperature window are conservative with respect to similar releases at the low end of the feedwater temperature window. At the high end, there is more energy available for release into containment.

3.2.2.4 Auxiliary Feedwater System

Generally, within the first minute following a steamline break, the auxiliary feedwater (AFW) system is initiated on any one of several protection system signals. Addition of auxiliary feedwater to the steam generators will increase the secondary mass available for release to containment as well as increase the heat transferred to the secondary fluid. The auxiliary feedwater flow to the faulted and intact steam generators has been assumed to be a function of the backpressure on the AFW pumps as a result of the depressurizing steam generator in the steamline break analysis inside containment. A range of cavitating venturi sizes in each of the AFW supply lines to the steam generators has been assumed that maximizes flow to the faulted-loop steam generator and minimizes flow to the intact-loop steam generators. The volume of the AFW piping is minimized. Purging of AFW piping is not assumed since a minimum volume permits colder AFW to be injected into the steam generator rather than any hotter auxiliary feedwater resident in the piping. The more dense injected AFW causes a greater mass addition to the faulted-loop steam generator than if the resident auxiliary feedwater had to be purged prior to the flow of AFW into the steam generator. Auxiliary feedwater flow to the faulted-loop steam generator has been assumed up until the time of operator action at 30 minutes after event initiation to isolate the flow to the steam generator near the break location. Auxiliary feedwater system assumptions that have been used in the analysis are presented in Table 3.2-3.

3.2.2.5 Steam Generator Fluid Mass

A maximum initial steam generator mass in the faulted-loop steam generator has been used in all of the analyzed cases. The use of a high faulted-loop initial steam generator mass maximizes the steam generator inventory available for release to containment. The initial mass has been calculated as the value corresponding to the programmed level +7 percent narrow-range span and assuming 0 percent tube plugging, plus a mass uncertainty. This assumption is conservative with respect to the RCS cooldown through the faulted-loop steam generator resulting from the steamline break.

3.2.2.6 Steam Generator Reverse Heat Transfer

Once the steamline isolation is complete, the steam generators in the intact loops may become sources of energy that can be transferred to the steam generator with the broken steamline. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes could drop below the temperature of the secondary fluid in the intact steam generators, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steamline. When applicable, the effects of reverse steam generator heat transfer are included in the results.

3.2.2.7 Break Flow Model

Piping discharge resistances are not included in the calculation of the releases resulting from the steamline ruptures [Moody Curve for an $f(\ell / D) = 0$ is used]. This is consistent with the expectations of the NRC as presented in Section 6.2.1.4 of the Standard Review Plan. For the BVPS-1 analysis, entrainment is assumed in the break effluent only for the large DERs upstream of the inline flow restrictor in the steamline. This is the same assumption as in the current steamline rupture analysis for this break size for BVPS-1 and has been approved for use regarding Model 51 steam generator designs as documented in References 1 and 2. For the DERs downstream of the flow restrictor for both Beaver Valley Power Station units, no entrainment is assumed in the break effluent. The assumption of saturated steam being released for this break location is a conservative assumption that maximizes the energy release into containment.

3.2.2.8 Steamline Volume Blowdown

The contribution to the mass and energy releases from the steam in the secondary plant main steam loop piping and header has been included in the mass and energy release calculations. The initial flowrate is determined using the Moody correlation, the pipe cross-sectional area, and the initial steam pressure. This blowdown is calculated only for the DER steamline break.

Separate calculations are performed whether or not there is an assumed failure of the MSCV for BVPS-1. A conservative steam piping volume of 7,876 ft^3 is used in this blowdown calculation, for a 1.4-ft^2 break, with an assumed MSCV failure, representing the main steam piping from the steam generator to the turbine throttle valve and to the intact-loop in-line flow restrictors. For the 4.6-ft^2 break, the in-line flow restrictor is in the same loop and a steam piping volume of 632 ft^3 is used in this blowdown calculation. The 6,654-ft^3 volume downstream of the in-line flow restrictor, bounded by the intact-loop MSIVs, is separately accounted for in the analysis at the time the intact-loop MSIVs close. When there is no assumed failure of the MSCV, a value of 930 ft^3 is used for the unisolable volume. A conservative steam piping volume of 7,192 ft^3 is used in the BVPS-2 blowdown calculation representing the main steam piping from the steam generator to the turbine throttle valve.

For the split-rupture steamline break, the unisolable steam mass in the piping is included as part of the initial inventory in the faulted-loop steam generator since the break is not large enough to cause a sudden decompression of the piping. The steamline break cases that do not assume a failure of the MSCV in the BVPS-1 faulted loop use a value of 930 ft^3 for the unisolable volume. The steamline break cases that assume a failure of the MSCV in the BVPS-1 faulted loop use a greater value (7,286-ft^3) for the unisolable volume post blowdown. The steamline break cases that do not assume a failure of the MSIV in the BVPS-2 faulted loop use a value of 1,038 ft^3 for the unisolable volume. The steamline break cases that assume a failure of the MSIV in the BVPS-2 faulted loop use a greater value (6,023-ft^3) for the unisolable volume post blowdown.

The analytical method of addressing the steamline piping blowdown and the effect of an MSCV (or MSIV) failure or no MSCV (or MSIV) failure is discussed in subsection 3.2.3.

3.2.2.9 Main Steamline Isolation

When there is an assumed MSCV failure in the BVPS-1 faulted steamline, steamline isolation is assumed in the unfaulted loops to terminate the blowdown from the two intact steam generators. A delay time of 8 seconds, accounting for delays associated with signal processing plus MSIV stroke time, with unrestricted steam flow through the valve during the valve stroke, has been assumed. When the MSCV on the faulted loop does not fail, it isolates the intact-loop steam generators from the break, and MSIV closure does not affect the accident progression.

For BVPS-2, steamline isolation is assumed in all three loops to terminate the blowdown from the two intact steam generators. A delay time of 7 seconds, accounting for delays associated with signal processing plus MSIV stroke time, with unrestricted steam flow through the valve during the valve stroke, has been assumed.

3.2.2.10 Reactor Trip System Actuations

The trip systems available to mitigate the effects of a MSLB accident inside containment include reactor trip, safety injection, steamline isolation, and feedwater isolation. The trip system actuation signals and associated setpoints that have been modeled in the analysis are identified in Table 3.2-4. The setpoints used are conservative values with respect to the reactor and containment protection setpoints proposed for the containment.

For the double-ended rupture MSLB at all power levels, the first trip system signal actuated is Low Steamline Pressure (lead/lag compensated in each channel) in any loop that initiates safety injection and steamline isolation (credited only for an assumed MSCV failure for BVPS-1); the safety injection signal produces a reactor trip signal. Feedwater system isolation occurs as a result of the safety injection signal.

For the split-rupture steamline breaks at all power levels, no mitigation signals are received from either the Reactor Trip System or any secondary-side signals produced by the Engineered Safety Features Actuation System. The first trip system signal actuated is assumed to be the High Containment Pressure, which initiates safety injection; the safety injection signal produces a reactor trip signal. Feedwater system isolation occurs as a result of the safety injection signal. Steamline isolation is initiated following receipt of the Intermediate-High-High Containment Pressure signal (credited only for an assumed MSCV failure for BVPS-1).

The turbine stop valve is assumed to close instantly following the reactor trip signal; the delay time used in the steamline break mass and energy releases inside containment is 0.0 seconds.

3.2.2.11 Safety Injection System

Minimum safety injection system (SIS) flowrates corresponding to the failure of one SIS train have been assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow is assumed to be 27 seconds for this analysis with offsite power available. A coincident loss of offsite power is not assumed for the analysis of the steamline break inside containment since the mass and energy releases would be reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

3.2.2.12 Reactor Coolant System Metal Heat Capacity

As the primary side of the plant cools, the temperature of the reactor coolant could drop below the temperature of the reactor coolant piping, the reactor vessel, the reactor coolant pumps, and the steam generator thick-metal mass and tubing. As this occurs, the heat stored in the metal is available to be transferred to the steam generator with the broken line. The effects of this RCS metal heat are included in the results using conservative thick-metal masses and heat transfer coefficients.

3.2.2.13 Core Decay Heat

Core decay heat generation assumed in calculating the steamline break mass and energy releases is based on the 1979 ANS Decay Heat + 2σ model (Reference 3). The existing analysis assumed the use of the 1971 standard (+20 percent uncertainty) for the decay heat. The assumption of using the 1979 version represents a deviation from the current licensing-basis analysis MSLB mass and energy release analysis for Beaver Valley Power Station.

3.2.2.14 Rod Control

The rod control system is conservatively assumed to be in manual operation for all steamline break analyses.

3.2.2.15 Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions are used to maximize the reactivity feedback effects resulting from the steamline break, consistent with the kinetics modeling described in Reference 1. Use of maximum reactivity feedback results in higher power generation if the reactor returns to criticality, thus maximizing heat transfer to the secondary side of the steam generators.

3.2.3 Description of the Analysis

The system transient that provides the break flows and enthalpies of the steam release through the steamline break inside containment has been analyzed with the LOFTRAN (Reference 4) computer code. Blowdown mass and energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick-metal heat storage including steam generator thick-metal mass and tubing, and reverse steam generator heat transfer. As noted in subsection 3.2.2.7, entrainment is assumed in the break effluent only for the large DERs upstream of the BVPS-1 inline flow restrictor in the steamline. For the DERs downstream of the BVPS-1 or BVPS-2 flow restrictor, no entrainment is assumed in the break effluent. The assumption of saturated steam being released for this break location is a conservative assumption that maximizes the energy release into containment. This reflects a conservative deviation in the Beaver Valley Power Station licensing basis since the current MSLB mass and energy releases analysis downstream of the flow restrictor assumes entrainment for the DER steamline break cases.

The existing MSLB mass and energy release analysis inside containment was performed using the MARVEL code as documented in WCAP-8822. The use of the LOFTRAN code for the analysis of the MSLB mass and energy releases is documented in Supplement 1 of WCAP-8822 (Reference 1) and has been reviewed and approved by the NRC for this application. The LOFTRAN code has been utilized previously for the Beaver Valley licensing-basis safety analyses.

The BVPS-1 and -2 NSSS have been analyzed to determine the transient steam mass and energy releases inside containment following a steamline break event. The approved methodology for the MSLB mass and energy releases inside containment as documented in Supplement 2 of WCAP-8822 (Reference 1) does not require that the effects of steam superheat be considered for input to an analysis assuming a large, dry containment. Since the Beaver Valley Power Station containment design is of this type, the steam superheat in the generator assumption has not been considered in this analysis for the steam generator replacement and EPU program. The resulting tables of mass and energy releases are used as input conditions to the analysis of the containment response.

3.2.3.1 BVPS-1 MSLB Analysis

The following licensing-basis cases of the MSLB inside containment have been analyzed at the noted conditions for the BVPS-1 EPU. As noted in subsection 3.2.2, all cases of the BVPS-1 steamline rupture (except Case 22M1) have assumed a single failure of the FIV in the faulted-loop main feedwater line. Any additional single failures are noted in the list below.

Case 1M: Large double-ended (4.6 ft^2) rupture at 100.6 percent power – MSCV failure

Case 2M: Large double-ended (4.6 ft^2) rupture at 100.6 percent power – no single failure in MSLB transient

Case 3M: Full double-ended (1.4 ft^2) rupture at 100.6 percent power – MSCV failure

Case 4M: Full double-ended (1.4 ft^2) rupture at 100.6 percent power – no single failure in MSLB transient

Case 5M: 0.725 ft^2 split rupture at 100.6 percent power – MSCV failure

Case 6M: 0.725 ft^2 split rupture at 100.6 percent power – no single failure in MSLB transient

Case 7M: Large double-ended (4.6 ft^2) rupture at 70 percent power – MSCV failure

Case 8M: Large double-ended (4.6 ft^2) rupture at 70 percent power – no single failure in MSLB transient

Case 9M: Full double-ended (1.4 ft^2) rupture at 70 percent power – MSCV failure

Case 10M: Full double-ended (1.4 ft^2) rupture at 70 percent power – no single failure in MSLB transient

Case 11M: 0.710 ft^2 split rupture at 70 percent power – MSCV failure

Case 12M: 0.710 ft^2 split rupture at 70 percent power – no single failure in MSLB transient

Case 13M: Large double-ended (4.6 ft^2) rupture at 30 percent power – MSCV failure

Case 14M: Large double-ended (4.6 ft^2) rupture at 30 percent power – no single failure in MSLB transient

Case 15M: Full double-ended (1.4 ft^2) rupture at 30 percent power – MSCV failure

Case 16M: Full double-ended (1.4 ft^2) rupture at 30 percent power – no single failure in MSLB transient

Case 17M: 0.424 ft^2 split rupture at 30 percent power – MSCV failure

Case 18M: 0.424 ft^2 split rupture at 30 percent power – no single failure in MSLB transient

Case 19M: Large double-ended (4.6 ft^2) rupture at 0 percent power – MSCV failure

Case 20M: Large double-ended (4.6 ft^2) rupture at 0 percent power – no single failure in MSLB transient

Case 21M: Full double-ended (1.4 ft^2) rupture at 0 percent power – MSCV failure

Case 22M: Full double-ended (1.4 ft^2) rupture at 0 percent power – FIV failure only

Case 22M1: Full double-ended (1.4 ft^2) rupture at 0 percent power – no single failure in MSLB transient

Case 23M: 0.209 ft^2 split rupture at 0 percent power – MSCV failure

Case 24M: 0.209 ft^2 split rupture at 0 percent power – no single failure in MSLB transient

For the large double-ended rupture cases upstream of the inline flow restrictor, the forward-flow cross-sectional area from the faulted-loop steam generator is limited by the area of the steam generator outlet nozzle, 4.6 ft^2. The reverse-flow area is limited by the cross-sectional flow area of the inline flow restrictor, 1.4 ft^2. However, the actual area of the main steam piping inside containment is 4.9 ft^2 for the section between the steam generator and the flow restrictor. Therefore, the larger cross-sectional area of the ruptured steamline expels steam faster than the smaller cross-sectional area of the flow restrictor can fill it. Thus, for the steamline rupture cases that assume an MSCV failure, the blowdown of the initial steam in the steamline piping of the faulted loop is modeled in the first fraction of a second of the event, followed by the reverse-flow blowdown from the intact-loop steam generators. The initial reverse-flow blowdown is discussed in subsection 3.2.2.8, Steamline Volume Blowdown, and provided in Table 3.2-5. The rate of the mass and energy releases from the steam header piping is a function of the initial pressure in the main steam system, which increases with decreasing power. At the time of MSIV closure, the steam flow from the intact-loop steam generators is terminated, but it is assumed that all steam that has exited the steam generator prior to steamline isolation is released through the break.

For the double-ended rupture cases downstream of the inline flow restrictor, the forward-flow cross-sectional area from the faulted-loop steam generator is limited by the flow restrictor area of 1.4 ft^2, which is less than the actual area of 4.9 ft^2 for the main steam piping inside containment. The cross-sectional area of the steam piping at this location is larger than the sum of the flow restrictors in the intact-loop steam generators. Therefore, the larger cross-sectional area of the ruptured steamline expels steam faster than the smaller cross-sectional area of the intact-loop steam generator flow restrictors can fill it. Thus, for the steamline rupture cases that assume an MSCV failure, the blowdown of the initial steam in the steamline header piping is modeled in the first few seconds of the event, followed by the reverse-flow blowdown from the intact-loop steam generators until MSIV closure in those loops. The initial reverse-flow blowdown is discussed in subsection 3.2.2.8, Steamline Volume Blowdown, and provided in Table 3.2-5. The rate of the mass and energy releases from the steam header piping is a function of the initial pressure in the main steam system, which increases with decreasing power. At the time of MSIV closure, the steam flow from the intact-loop steam generators is terminated, but it is assumed that all steam that has exited the steam generator prior to steamline isolation is released through the break.

The full DERs, either upstream or downstream of the inline flow restrictor, represent the breaks producing the highest mass flowrate from the faulted-loop steam generator. Smaller DER break sizes are represented by a reduction in the initial steam blowdown rate at the time of the break. Therefore, no other DER break sizes have been considered other than the full DERs.

For the split-break MSLB cases, the break area is smaller than the area of a single flow restrictor. Thus, for the steamline rupture cases that assume an MSCV failure, the flowrate from all steam generators prior to MSIV closure in the two unfaulted steamlines and the flowrate from a single steam generator after MSIV closure supply the steam flow to the break. The steam in the unisolable portion of the steamline does not affect the blowdown until the time of steam generator dry out, when the flowrate from the steam generator would decrease below the critical flowrate out of the break. At this point, the additional steam in the piping begins to have an effect on break flowrate until the steamline piping is empty. To model this effect in LOFTRAN, the mass of the unisolable steam in the steamline is added to the initial mass of the faulted steam generator. This accurately reflects both the total mass and energy that will be released from the break, and the timing of the effect of the unisolable steamline volume on the blowdown. When the MSCV in the faulted steamline is credited to close, the unisolable steamline volume is 930 ft^3. A failure of the MSCV on the faulted loop increases the unisolable steamline volume to a conservatively large value of 7,286 ft^3 for the post blowdown releases.

All the cross-sectional split-rupture areas have been redefined based on the assumption of operation with the Westinghouse-design Model 51 steam generators, at the uprated power, crediting the MSCVs, and with plant-specific values for the secondary-side protection system setpoints. Each break size as a function of power is the largest area that does not produce a steamline isolation signal from the Westinghouse SSPS, nor result in water entrainment in the break effluent as discussed in Reference 1. See subsection 3.2.2 for a discussion of the Model 54F replacement steam generators.

3.2.3.2 BVPS-2 MSLB Analysis

The following licensing-basis cases of the MSLB inside containment have been analyzed at the noted conditions for the BVPS-2 EPU program.

Case 1M: Full double-ended (1.069 ft^2) rupture at 100.6 percent power – MSIV failure

Case 2M: Full double-ended (1.069 ft^2) rupture at 100.6 percent power – FIV failure

Case 3M: 0.753 ft^2 split rupture at 100.6 percent power – no single failure in MSLB transient

Case 4M: 0.753 ft^2 split rupture at 100.6 percent power – MSIV failure

Case 5M: 0.753 ft^2 split rupture at 100.6 percent power – FIV failure

Case 6M: Full double-ended (1.069 ft^2) rupture at 70 percent power – MSIV failure

Case 7M: Full double-ended (1.069 ft^2) rupture at 70 percent power – FIV failure

Case 8M: 0.757 ft^2 split rupture at 70 percent power – no single failure in MSLB transient

Case 9M: 0.757 ft^2 split rupture at 70 percent power – MSIV failure

Case 10M: 0.757 ft^2 split rupture at 70 percent power – FIV failure

Case 11M: Full double-ended (1.069 ft^2) rupture at 30 percent power – MSIV failure

Case 12M: Full double-ended (1.069 ft^2) rupture at 30 percent power – FIV failure

Case 13M: 0.756 ft^2 split rupture at 30 percent power – no single failure in MSLB transient

Case 14M: 0.756 ft^2 split rupture at 30 percent power – MSIV failure

Case 15M: 0.756 ft^2 split rupture at 30 percent power – FIV failure

Case 16M: Full double-ended (1.069 ft^2) rupture at 0 percent power – MSIV failure

Case 17M: Full double-ended (1.069 ft^2) rupture at 0 percent power – FIV failure

Case 18M: 0.608 ft^2 split rupture at 0 percent power – no single failure in MSLB transient

Case 19M: 0.608 ft^2 split rupture at 0 percent power – MSIV failure

Case 20M: 0.608 ft^2 split rupture at 0 percent power – FIV failure

For the double-ended rupture cases, the forward-flow cross-sectional area from the faulted-loop steam generator is limited by the integral flow restrictor area of 1.069 ft^2, which is less than the actual area of 4.9 ft^2 for the main steam piping inside containment. The cross-sectional area of the steam piping at this location is larger than the sum of the flow restrictors in the intact-loop steam generators. Therefore, the larger cross-sectional area of the ruptured steamline expels steam faster than the smaller cross-sectional area of the intact-loop steam generator flow restrictors can fill it. Thus, the blowdown of the initial steam in the steamline header piping is modeled in the first few seconds of the event, followed by the reverse-flow blowdown from the intact-loop steam generators until MSIV closure. The initial reverse-flow blowdown is discussed in subsection 3.2.2.8, Steamline Volume Blowdown, and provided in Table 3.2-6. The rate of the mass and energy releases from the steam header piping is a function of the initial pressure in the main steam system, which increases with decreasing power. At the time of MSIV closure, the steam flow from the intact-loop steam generators is terminated, but it is assumed that all steam that has exited the steam generator prior to steamline isolation is released through the break. This is consistent with the Reference-1 methodology, and means that there is not a differentiation of the effect of an MSIV failure for the full DER MSLB cases.

The full DER represents the break producing the highest mass flowrate from the faulted-loop steam generator. Smaller DER break sizes are represented by a reduction in the initial steam blowdown rate at the time of the break. Therefore, no other DER break sizes have been considered other than the full DER.

For the split-break MSLB cases, the break area is smaller than the area of a single integral flow restrictor. The flowrate from all steam generators prior to MSIV closure and the flowrate from a single steam generator after MSIV closure supply the steam flow to the break. The steam in the unisolable portion of the steamline does not affect the blowdown until the time of steam generator dry out, when the flowrate from the steam generator would decrease below the critical flowrate out of the break. At this point, the additional steam in the piping begins to have an effect on break flowrate until the steamline piping is empty. To model this effect in LOFTRAN, the mass of the unisolable steam in the steamline is added to the initial mass of the faulted steam generator. This accurately reflects both the total mass and energy that will be released from the break, and the timing of the effect of the unisolable steamline volume on the blowdown. When all MSIVs are credited to successfully close, the unisolable steamline volume is 1,038 ft^3. A failure of the MSIV on the faulted loop increases the unisolable steamline volume to a conservatively large value of 6,023 ft^3 for the post blowdown releases.

All the cross-sectional split-rupture areas have been redefined based on the assumption of operation with the Westinghouse-design Model 51M steam generators, at the uprated power, with plant specific values for the secondary-side protection system setpoints. Each break size as a function of power is the largest area that does not produce a steamline isolation signal from the Westinghouse Solid State Protection System (SSPS), nor result in water entrainment in the break effluent as discussed in Reference 1.

3.2.4 Acceptance Criteria

The main steamline break is classified as an ANS Condition IV event, an infrequent fault. The acceptance criteria associated with the steamline break event resulting in a mass and energy release inside containment is based on an analysis that provides sufficient conservatism to show that the containment design margin is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model, main and auxiliary feedwater flow, steamline and feedwater isolation, and single failure such that the containment peak pressure and temperature are maximized. These analysis assumptions have been included in this steamline break mass and energy release analysis as discussed in Reference 1 and subsection 3.2.2 of this report.

3.2.5 Results

Using Reference 1 as a basis, including parameter changes associated with the power uprating, the mass and energy release rates for each of the steamline break cases noted in subsection 3.2.3 have been developed for use in containment pressure and temperature response analyses.

The only major event other than the MSLB that results in a breach of the secondary-side piping is the MFLB. Main Feedwater Line Breaks (MFLB) are not analyzed since such break scenarios result in a blowdown less limiting than the MSLB because the pipe break mass flow for the MFLB is limited by the steam generator internals design. The feedwater enthalpy at any power is less than the enthalpy of saturated steam at the secondary-side operating pressures. Therefore, the long-term integrated energy released following a MFLB is bounded by the long-term integrated energy released following a MSLB. The containment pressure and temperature responses to the mass and energy releases from a MFLB are bounded by the containment responses following the MSLB event.

3.2.6 Conclusions

The mass and energy releases from the 25 steamline break cases for BVPS-1 and the 20 steamline break cases for BVPS-2 have been analyzed at the conditions defined by the uprated power level. The assumptions delineated in subsection 3.2.2 have been included in the steamline break analysis such that the results are consistent with and continue to comply with the current Beaver Valley licensing-basis/acceptance requirements. The steam mass and energy releases discussed in this section have been provided for use in the containment response analysis in support of the Beaver Valley Power Station EPU program.

3.2.7 References

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

2. WCAP-8821-P-A (Proprietary) and WCAP-8859-A (Nonproprietary), "TRANFLO Steam Generator Code Description," September 1976; approved version June 2001.

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

4. WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), "LOFTRAN Code Description," April 1984.

Table 3.2-1
Beaver Valley Power Station

Nominal Plant Parameters for Extended Power Uprate*
(MSLB Mass and Energy Releases Inside Containment)

Nominal Conditions	
NSSS Power, MWt	2910
Reactor Coolant Pump Heat, MWt	15**
Reactor Coolant Flow (total), gpm (Thermal Design Flow)	261,600
Pressurizer Pressure, psia	2250
Core Bypass, %	6.5
Reactor Coolant Vessel Average Temperature, °F	580.0
Steam Generator***	
Steam Temperature, °F	521.9
Steam Pressure, psia	826
Steam Flow, 10^6 lbm/hr (Plant Total)	13.05
Feedwater Temperature, °F	455
Zero-Load Temperature, °F	547

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window.

** Maximum reactor coolant pump heat used in the MSLB mass and energy releases.

*** Steam generator performance data used in the analysis is conservatively high for steam temperature and pressure.

Table 3.2-2
Beaver Valley Power Station

Initial Condition Assumptions for Extended Power Uprate*

MSLB Mass and Energy Releases Inside Containment

Initial Conditions	**Power Level (%)**			
Parameter	**100.6**	**70**	**30**	**0**
RCS Average Temperature (°F)	588.5	578.6	565.4	547.0
RCS Flowrate (gpm) (Thermal Design Flow)	261,600	261,600	261,600	261,600
RCS Pressure (psia)	2250	2250	2250	2250
Pressurizer Water Volume (ft^3)	834.3	693.3	505.3	411.3 (U1) 364.3 (U2)
Feedwater Enthalpy (Btu/lbm)	436.0	385.4	305.3	70.7
SG Pressure (psia)**	885	935	1011	1004
SG Water Level (% NRS)	51	51	51	51

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window; temperatures include applicable calorimetric uncertainties.

** The noted SG pressures are determined at the steady-state conditions defined by the RCS average temperatures, including applicable uncertainties except at 0% power.

Table 3.2-3
Beaver Valley Power Station

Main and Auxiliary Feedwater System Assumptions for Extended Power Uprate

MSLB Mass and Energy Releases Inside Containment	
Main Feedwater System	
Flowrate – DERs @ all powers (until main feedwater isolation)	Feedwater flow based on system performance as a function of SG pressure.
Flowrate – split ruptures @ all powers (until main feedwater isolation)	Feedwater flow matches steam flow.
Unisolable volume from SG nozzle to FRV assuming a single failure of the FIV (faulted loop) (BVPS-1)	268 ft^3
Unisolable volume from SG nozzle to FIV (faulted loop) (BVPS-2)	157 ft^3
Unisolable volume from SG nozzle to FCV assuming a single failure of the FIV (faulted loop) (BVPS-2)	264 ft^3
Auxiliary Feedwater System	
Flowrate to all steam generators	Maximum flow to each SG is 310 gpm. The actual data used is a function of SG pressure.
Temperature (maximum value)	120°F
Piping purge volume (faulted loop)	1 ft^3
Actuation delay time	0 seconds

Table 3.2-4
Beaver Valley Power Station

Protection System Actuation Signals and Safety System Setpoints for Power Uprate

MSLB Mass and Energy Releases Inside Containment	
Reactor Trip 2/3 Low Pressurizer Pressure – 1935 psia Safety Injection	
Safety Injection 2/3 Low Pressurizer Pressure – 1745 psia (BVPS-1) 1760 psia (BVPS-2) 2/3 Low Steamline Pressure in any loop – 460 psia dynamic compensation lead - 50 seconds lag - 5 seconds 2/3 High Containment Pressure – 22 psia	
Steamline Isolation 2/3 Low Steamline Pressure in any loop – 460 psia dynamic compensation lead - 50 seconds lag - 5 seconds 2/3 Intermediate High-High Containment Pressure – 24 psia	(only credited in analyses that assume an MSCV single failure for BVPS-1)
Feedwater Isolation and Auxiliary Feedwater Initiation Safety Injection	

Table 3.2-5
Beaver Valley Power Station (BVPS-1)

Mass and Energy Flowrates for Steam Piping Reverse Flow Blowdown – Applicable to the DER MSLBs Inside Containment with an Assumed MSCV Single Failure

Power Level	100.6%	70%	30%	0%
Steam Mass Flowrate (lbm/sec)	8,947.4	9,483.0	10,137.1	10,211.1
Steam Energy Flowrate (10^6 Btu/sec)	10.709	11.334	12.088	12.180
Duration of Blowdown (sec) for the Large (4.6 ft^2) Break	0.138	0.139	0.142	0.139
Duration of Blowdown (sec) for the Full (1.4 ft^2) Break	1.725	1.729	1.764	1.737

Table 3.2-6
Beaver Valley Power Station (BVPS-2)

Mass and Energy Flowrates for Steam Piping Reverse Flow Blowdown – Applicable to the DER MSLBs Inside Containment

Power Level	100.6%	70%	30%	0%
Steam Mass Flowrate (lbm/sec)	8,957.2	9,481.5	10,285.6	10,211.1
Steam Energy Flowrate (10^6 Btu/sec)	10.721	11.332	12.266	12.180
Duration of Blowdown (sec)	1.58	1.58	1.59	1.59

3.3 SHORT-TERM MASS AND ENERGY RELEASES (LOCA)

The approved Westinghouse methodology for the short term LOCA mass and energy release analysis is documented in "Topical Report Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, Rev.1, August 1975 (Proprietary), WCAP-8312-A, Rev. 2 (Non-Proprietary) (Reference 1). The critical flow calculation employs appropriately defined critical flow correlations applied for fluid conditions at the break location. For the early portion of blowdown, subcooled, saturated and two phase critical flow regimes are encountered. The WCAP-8264 methodology uses the Moody correlation for saturated and two-phase conditions and a modification of the Zaloudek correlation for the subcooled blowdown regime. The details of these models and comparisons to other models and experimental data are described in detail in WCAP-8264. Most short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition. The Zaloudek correlation, which models this condition, is currently used in the short-term LOCA mass and energy release analyses. The analysis input that has the potential to change with this program is the initial RCS fluid temperature. The use of lower temperatures maximizes the critical mass flux in the Zaloudek correlation. Since this event lasts for approximately 3 seconds, the single effect of power and various steam generator designs are not significant.

3.3.1 BVPS-1 Discussion and Conclusion

For BVPS-1, based upon LBB (References 2-3), the dynamic effects associated with RCS primary pipe breaks and the pressurizer surge line break do not have to be included in the licensing analysis. Therefore, the current releases associated with these breaks have been eliminated from consideration. The benefits of reduced mass and energy releases associated with the smaller RCS branch lines, as compared to the larger RCS primary breaks and surge line break, more than offset any penalties associated with possible increased releases, which could result from decreased RCS coolant temperatures associated with the NSSS EPU. However, the evaluation determined that the existing BVPS-1 short-term mass and energy releases for the spray line break used for the pressurizer superstructure are increased due to the RCS temperature changes associated with the NSSS EPU. Table 3.3-1 presents the releases for the NSSS EPU.

3.3.2 BVPS-2 Discussion and Conclusion

For BVPS-2, based upon LBB (References 4, 5 and 6), the dynamic effects associated RCS pipe breaks, including RCS nozzle breaks down to and including 6 inch piping, do not have to be included for licensing analysis. Therefore, the current releases associated with the Surge Line Break, RCS split break at the steam generator inlet elbow, and the 150 in^2 RCS cold leg break have been eliminated from consideration. The benefits of the decrease in mass and energy releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS and larger RCS nozzle pipe breaks, more than offsets any penalties associated with possible increased releases which will result from decreased RCS coolant temperatures associated with the NSSS EPU. Also, based upon a comprehensive evaluation of the pressurizer spray line at BVPS-2, the current BVPS-2 UFSAR spray line mass and energy releases are bounding for the NSSS EPU without the need to adjust for lower RCS temperatures or higher RCS pressures.

3.3.3 References

1. "Topical Report Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, Rev.1 , August 1975 (Proprietary), WCAP-8312-A , Rev.2 (Non-Proprietary).

2. "Beaver Valley Unit 1 – Removal of Large-Bore Snubbers from Primary Coolant Loops (TAC 65107)," from Mr. Peter S. Tam, Project Manager, Project Directorate I-4, Division of Reactor Projects I/II, United States Nuclear Regulatory Commission to Mr. J. D. Sieber, Vice President Nuclear Operation, Duquesne Light Company, December 9, 1997.

3. "Approval of Leak-Before-Break Analysis (TAC No. 72110)," NRC Letter from Albert W. DeAgazio, Sr, Project Manager, Project Directorate I-4, Division of Reactor Projects - I/II, Office of Nuclear Reactor Regulation to Mr. J. D. Sieber, Vice President Nuclear Group, Duquesne Light Company, May 2, 1991.

4. "Safety Evaluation Report related to the operation of Beaver Valley Power Station, BVPS-2 Docket No. 50-412 Duquesne Light Company, et al.," NUREG-1057, Supplement No. 4, March 1987.

5. "Safety Evaluation Report related to the operation of Beaver Valley Power Station, BVPS-2 Docket No. 50-412 Duquesne Light Company, et al.," NUREG-1057, Supplement No. 5, May 1987.

6. "Elimination of Dynamic Effects of Postulated Pressurizer Surge Line Rupture and Elimination of Reactor Coolant System Component Support Snubbers (TAC NOS. 71086 and 72111)," from Mr. Albert W. DeAgazio, Sr. Project Manager, Project Directorate I-4, Division of Reactor Projects –I/II, Office of Nuclear Reactor Regulation to Mr. J. D. Sieber, Vice President Nuclear Group, Duquesne Light Company, April 8, 1991.

Table 3.3-1
Mass and Energy Release Rates for DER of Spray Line in Pressurizer Subcompartment Superstructure (Above El. 767 Ft-10 Inches)

Time (Seconds)	Mass Flow[1] (Lbm/sec)	Energy Release[2] (Btu/second)
0.0	0.0	0.0
0.02501	2,572.7	1,545,186.3
0.05005	2,561.0	1,537,966.6
0.07509	2,561.0	1,537,893.6
0.1001	2,526.0	1,518,577.2
0.1250	2,540.0	1,526,742.5
0.1500	2,527.2	1,518,747.0
0.1749	2,519.0	1,515,031.5
0.2001	2,513.2	1,511,293.5
0.2249	2,491.0	1,500,485.2
0.2500	2,501.5	1,505,269.8
0.2750	2,486.3	1,497,847.7
0.3000	2,475.8	1,492,722.5
0.3249	2,463.0	1,486,386.3
0.3501	2,460.6	1,486,027.7
0.3750	2,447.8	1,481,012.6
0.4000	2,442.0	1,478,417.8
0.4250	2,435.0	1,473,536.5
0.4499	2,421.0	1,469,923.2
0.4749	2,421.0	1,469,661.3
0.5000	2,403.5	1,463,068.8
0.6500	2,367.3	1,448,110.0
0.8000	2,344.0	1,437,356.8
1.0000	2,318.3	1,423,449.1
1.2000	2,292.6	1,409,429.0
1.3750	2,272.8	1,397,984.3
1.5000	2,260.0	1,390,809.6
1.7000	2,242.5	1,380,435.3

Table 3.3-1 (Continued)
Mass and Energy Release Rates for DER of Spray Line in Pressurizer Subcompartment Superstructure (Above El. 767 Ft-10 Inches)

Time (Seconds)	Mass Flow[1] (Lbm/sec)	Energy Release[2] (Btu/second)
1.7999	2,234.3	1,375,635.0
1.8999	2,226.1	1,371,054.8
1.9750	2,221.5	1,367,968.8
2.0999	2,213.3	1,362,906.5
2.2000	2,208.6	1,358,970.6
2.2999	2,200.5	1,355,275.3
2.4000	2,195.8	1,351,673.3
2.5000	2,190.0	1,348,053.3
2.5999	2,185.3	1,344,444.6
2.7000	2,179.5	1,340,898.7
3.0000	2,163.1	1,329,747.6

Note:

1,2 These mass and energy releases have been modified to reflect the extended power uprate.

3.4 MSLB MASS AND ENERGY RELEASES OUTSIDE CONTAINMENT

The analysis of the MSLB mass and energy releases outside containment assumes minimum flowrates from the AFW system, with preferential flow to the two steam generators in the unfaulted steamlines. Minimum AFW flowrates are used to develop a conservative model with respect to early uncovery of the tubes in the faulted-loop steam generator and early generation of superheated steam for release outside containment. The installation of the cavitating venturis in the AFW system limits the maximum flow to any single steam generator; minimum flowrates are also affected by the venturis. The MSLB mass and energy releases outside containment for BVPS-1 related to the AFW flowrates are affected by the installation of the cavitating venturis. The revised releases were evaluated in Section 7.11.2.

The MSLB mass and energy releases outside containment for BVPS-2 are not affected by the containment conversion.

3.5 RADIOLOGICAL ANALYSIS RELEASES

An evaluation was performed for the Control Rod Ejection Accident (CREA) in order to determine the resulting environmental steam releases. These releases were provided as input for use in the radiological dose analysis (Chapter 5).

The evaluation is based upon the same parameters and assumptions as used for the SBLOCA analysis including the uprated power level and uses a conservative bounding approach in order to provide conservative steam releases for use in the radiological dose analysis.

3.5.1 Results

The evaluation of the CREA resulted in the following input to the radiological analysis:

- Safety injection signal generated on low pressurizer pressure < 75 seconds,
- Primary pressure falls below secondary pressure < 2500 seconds,
- Steam release from secondary system stops ≤ 2500 seconds,
- Total steam flows assuming reactor trip at the start of the event:

0 - 150 seconds	≤ 900lb/sec
150 - 300 seconds	≤ 300 lb/sec
300 - 2500 seconds	≤ 150 lb/sec
> 2500 seconds	0.0 lb/sec

It has also been determined that the environmental steam releases provided for the Locked Rotor and Loss of AC Power events can be used to represent the steam release after 2500 seconds to RHR initiation. The Locked Rotor releases assume that the cooldown to RHR conditions is initiated at 2 hours, which also would apply to the CREA.

The CREA steam releases have been developed for BVPS-1 and -2 at EPU conditions.

4 CONTAINMENT ANALYSIS

4.1 INTRODUCTION

The results of the containment response assessments performed to support conversion of Beaver Valley Power Station Unit 1 (BVPS-1) and Beaver Valley Power Station (BVPS-2) from sub-atmospheric to atmospheric design are provided in this chapter. These containment assessments incorporate the applicable proposed methodology changes, Technical Specification changes, and plant modifications as delineated in Chapter 1. In particular the atmospheric containment responses were quantified based upon the implementation of an Extended Power Uprate to a reactor core power level of 2900 MWt on BVPS-1 and –2 and upon the installation of replacement steam generators on BVPS-1.

Design bases events included in the containment analyses are the rupture of a pipe in the Reactor Coolant System (LOCA) and the Main Steam Line Break (MSLB) between the top of the steam generator and the penetration through the containment wall. LOCA breaks were evaluated at hot leg, cold leg, and pump suction locations. Each design bases event is addressed for the atmospheric containment configuration with the reactor operating at future uprated power. Evaluations for the limiting containment design basis events were evaluated to assess the following attributes:

- peak containment pressure,
- peak containment gas temperature,
- long term temperatures within the containment,
- peak liner temperatures,
- available Net Positive Suction Head (NPSH), and
- maximum sump water and recirculation spray heat exchanger cooling water outlet temperatures.

An additional evaluation was performed to verify that inadvertent actuation of the containment spray system during normal plant operation would not result in a containment pressure less than the minimum containment design pressure of 8 psia.

The evaluation of each of the above attributes depends on many design input parameters such as containment volume, pump flow rates, spray water temperature, heat transfer coefficients, and pump start times. Each of those design input parameters can itself vary through a range of values. Depending on the attribute being evaluated, the most conservative value within that range was identified and used. For example, consider the initial containment pressure design input parameter. To calculate the most conservative NPSH, the minimum value from the range of possible initial containment pressures is assumed. Whereas, to calculate the most conservative peak containment pressure, the maximum value from the range of possible initial containment pressures is assumed. The design input parameters and which value (max or min) was assumed for the calculation of each of the above attributes are discussed in Section 4. 3.

The mass and energy released to the containment can also vary depending upon a combination of variables such as break size, break location, single active failure, power level, and containment air pressure at the time of the break. The consequences of the breaks can further vary dependent on a variety of possible single active failures that may occur concurrent with the breaks and affect the availability of engineered safety features (ESFs). Single active failures that were considered to identify the "worst single failure" that maximizes the challenge to the containment integrity include:

- the failure of a single train of engineered safety features such as might occur with the failure of a Diesel Generator (DG) coincident with a loss of off-site power,
- the failure of a single train of low head safety injection (LHSI) (BVPS-1 only),
- the single failure of the containment isolation phase B signal (CIB), which would result in the failure of one complete train of quench and recirculation sprays to start, which means that the remaining train of sprays would be available to cool the containment atmosphere,
- the failure of a service water pump to supply cooling water to one train of the recirculation spray heat exchangers (two heat exchangers) which are part of the containment heat removal system,
- the failure of a main steam isolation valve (MSLB only),
- the failure of a main feedwater isolation valve (MSLB only),
- the failure of a main steam line check valve (MSLB only),
- the failure of a timer start relay which would result in the failure of one train of recirculation spray, and
- the failure of one train of quench spray.

A matrix was developed for each BVPS unit to ensure the most conservative cases were identified and evaluated. These matrices are provided in Tables 4-1 and 4-2 as discussed in Section 4. 3.

Furthermore, operational conditions in the reactor and steam generators also need to be examined for the worst possible conditions that could influence the mass and energy releases from that component experiencing the break. Hence, the MSLB analyses need to consider the potential for additional water inventory in the steam generators at reactor power levels less than 100%. Section 3.2.1 discusses the spectrum of MSLB mass and energy releases used to address the range of operational conditions.

Thus, the containment analyses were performed in a manner that ensured that the evaluations identified and examined the most severe challenges to successful operation of the containment and its supporting mechanical and electrical safety systems.

To accomplish this, mass and energy releases were generated using Westinghouse codes previously approved by the Nuclear Regulatory Commission (NRC). (See Chapter 3 for discussion of mass and energy release models.) The mass and energy releases were then input to the containment analysis module of the MAAP-DBA code which calculated the containment pressure and temperature attributes identified above. The pressure and temperature attributes calculated were less than the design pressure and temperature of the containment. Hence, structural re-analysis of the containment shell was unnecessary. Similarly, the pressure and temperature calculated were within the capabilities of the supporting mechanical and electrical systems, as discussed in Chapters 6 and 7.

4.1.1 Application of MAAP-DBA to Containment Assessments

The MAAP-DBA code was developed to allow the calculation of containment response attributes for a spectrum of postulated LOCA and main steamline break sequences as part of design basis calculations for BVPS-1 and –2 containments. The containment assessments for design basis application are implemented in a manner consistent with the NRC guidance provided in the Standard Review Plan. This includes the use of Tagami and Uchida heat transfer correlations for the quantification of the passive heat sink responses. The spectrum of containment response attributes to be quantified include the peak containment pressure, the short and long-term containment temperature, the containment liner temperature, the long-term sump water temperature, the available NPSH for ECCS and containment spray pumps, and the maximum service water outlet temperature for the containment heat removal heat exchanger. To address this set of containment response attributes for the spectrum of loss of coolant accident break sizes, both single node and multiple node containment models are used. The single node models apply for those design basis sequences and attributes that employ the Tagami and Uchida heat transfer correlations. For the multiple node applications, a heat and mass transfer analogy based on natural convection is used.

A single node model is used to calculate peak containment pressure and containment liner temperature as well as post accident containment global gas temperature profiles for equipment qualification. A multi-node model is used for NPSH and sump water temperature. This provides improved accountability of water hold up for NPSH and debris transport calculations.

The MAAP-DBA code and supporting tests, benchmarks and qualifications are described in the BVPS Pre Application Report (Reference 5) and in Chapter 9 of this report.

This section describes how the MAAP-DBA Generalized Containment Model (GCM) was applied to the BVPS plant specific analysis.

The Modular Accident Analysis Program (MAAP) is an integral representation of the reactor core, Reactor Coolant System (RCS) and containment. One aspect of the MAAP code structure is that the various major components, i.e., the core, the RCS and the containment are in essence separate major modules that can be used individually. Consequently, MAAP can generate its own mass and energy releases for use in its containment analysis model or it can use externally generated mass and energy releases.

As discussed in Chapter 3 all design bases LOCA and MSLB containment analysis for BVPS used mass and energy releases generated by NRC approved Westinghouse methodologies. The SBLOCA and IBLOCA containment analysis for BVPS used mass and energy releases generated by MAAP-DBA. The MAAP-DBA code used to generate those SBLOCA and IBLOCA mass and energy releases was benchmarked against NOTRUMP. Containment analysis results based on a model consistent with MAAP-DBA also generated SBLOCA mass and energies which were accepted by the NRC at the D. C. Cook plant.

4.1.1.1 Parameter File/Nodalization

The application of the MAAP-DBA containment model to a commercial nuclear power plant begins with the characterization of the containment building geometry, emergency safeguard systems, etc., in a plant specific parameter file. While this parameter file includes specifications for the entire plant (including the reactor coolant system), the application of the MAAP-DBA containment model to the large LOCA and MSLB design basis evaluations, with the external specification of the mass and energy releases into the containment, only requires that the containment information be qualified for the specific plant. The remaining information has been developed for the BVPS RCS designs such that it can also be used for medium and small break LOCA evaluations.

When formulating a containment parameter file, the most important decision lies in the specification of the number of nodes used to represent the building. To be consistent with the previous BVPS DBA analyses, the evaluations for peak pressure and temperature are performed using single node models. However, those evaluations which are sensitive to potential water accumulation (holdup) in various locations within the building are performed with multi-node models, i.e., 18 nodes for BVPS-1 and 17 nodes for BVPS-2.

There are a few guidelines to be followed for multi-node models.

1. Each building region which is a separate room or compartment with limited connections (flow paths) to the remainder of the building should be treated as a separate node. For example, the reactor cavity in a typical PWR large dry containment is generally separated from the remainder of the building by a thick concrete biological shield. Furthermore, the walls around the in-core instrument tubes that penetrate through the bottom of the reactor vessel in most PWR designs segregate the region from other compartments. Hence, this region should be one of the nodes. Furthermore, specific rooms such as the in-core instrumentation seal table room that may also be compartmentalized for shielding purposes should also be a separate node.

2. Typically the design basis accident conditions include analyses for a large break RCS LOCA as well as evaluations for a main steam line break. For those accident analyses requiring a multi-node model such as maximum recirculation sump temperature following a large break LOCA, the containment nodalization should include the region surrounding the reactor coolant system, the loop compartment(s), and the region above the operating deck as individual nodes. In this regard, the LOCA conditions considered include any sensitivities related to whether the LOCA is postulated to occur in any of the reactor coolant loops. Consequently, if the reactor coolant loops are in one large compartment, a single node is sufficient. Conversely, if the loop compartments and other RCS components, such as the pressurizer, are in individual rooms, then the nodalization scheme should be expanded to include each of these compartments as a separate node.

3. An important parameter of the DBA evaluations is the sump temperature under accident conditions. Thus, that region in the bottom of the containment which includes the recirculation sump and the floor of the containment outside the reactor cavity should be considered as a separate node.

4. The nodalization scheme needs to be sufficient to represent the potential for light gas stratification in the top of the containment building. Consequently, there should be at least two nodes (one above the other) in the region above the operating deck where light gases, such as steam could accumulate. This is the only region of the BVPS containment model that uses multiple nodes to represent an open region.

4.1.1.2 Flow Paths/Junctions

Multi-node models also require specification of the junctions (flow paths) connecting the various nodes. These are defined in the TOPOLOGY section of the parameter file. These junctions include doorways, hatchways, open areas, grating, etc. These junctions enable the major flow transport paths to be clearly specified and quantified with respect to their available area, their potential to be flooded by water accumulation, the potential for water accumulation within containment nodes, etc. Hence, this topology description is important in providing a realistic multi-node characterization, including the potential for global and countercurrent natural circulation, of the containment response to DBA conditions.

4.1.1.3 Structural Heat Sinks

Structural heat sink information including the surface areas, thicknesses, materials, whether they are steel lined, whether the outer surface is painted, etc., is also described in the parameter file. During DBA conditions the heat sink response is typically sufficiently slow that only a few heat sinks have the thermal conduction developed through the entire width of the heat sink. Nonetheless the MAAP-DBA parameter file has the capability for all of these heat sinks to be identified as two-sided structures, thereby enabling the parameter file to be used for DBA evaluations as well as for accident analyses evaluations over an extended time period, i.e., hours or days. To accomplish this, the node facing each heat sink surface is identified in the parameter file, i.e., a heat sink face is pointed to the specific node with which it interacts, and its opposite face is pointed to another node.

4.1.1.4 Engineered Safeguards

Engineered safeguards that are specific to the containment are also defined in the parameter file, including the containment spray pumps, and the heat exchangers that are used to remove decay heat from the containment during recirculation. The configuration of the ECCS and containment spray injection pumps must be specified in terms of:

- those pumps which take suction only from the Refueling Water Storage Tank (RWST),
- those pumps which take suction from the RWST and are switched over to take suction from the containment sump at containment recirculation,
- those pumps which only take suction from the containment sump under recirculation conditions.

Heat removal capabilities (if any) must be identified with the type of pumping system. Loss of function related to the single failure criterion are addressed in the input decks assembled for each sequence. The parameter file is meant to represent the nominal operating condition for specific systems. As part of this, the configuration also defines whether any pumps are "piggybacked" to the discharge of a lower pressure pump to increase their discharge pressure.

4.1.2 Treatment of the Mass and Energy Releases

As discussed in Section 3, there are a number of DBA accident conditions that need to be analyzed for the containment response including large break LOCA accident initiators as well as MSLB conditions. The discharge from the break location is the mass and energy source that is input into the containment atmosphere, and as a result, these must be derived from design basis accident analysis tools. To ensure that the spectrum of conditions for a specific plant have been addressed, a matrix of those accident sequences to be examined needs to be defined to ensure that the design basis attributes of the containment have been properly evaluated. These include:

- peak containment pressure,
- peak containment gas temperatures for Equipment Qualification (EQ),
- minimum available NPSH for all pumps that operate under recirculation conditions,
- maximum containment liner temperature,
- the maximum piping temperatures related to recirculation, cold leg injection and hot leg injection after switchover to this long term injection configuration, and
- the maximum service water temperature for heat removal during recirculation conditions.

Since each of these has a specific set of conditions, which combined with the single failure criterion provides the most challenging response to the attributes of interest, it is not surprising that each of these requires a separate analysis. For instance, the evaluations for the maximum temperatures within containment following an accident focus on those set of conditions which result in the hottest steam being released to the containment atmosphere, i.e., a double-ended break where the mass and energy streams from the two sides of the break (the hot water flow rate from the cold leg side and the steam flow from the steam generator side) are discharged into the containment atmosphere as separate streams. Conversely, the evaluations for the minimum available NPSH focus on those conditions which could result in the maximum sump temperature and the largest recirculation flow rate to maximize the frictional losses. In this case, the mass and energy releases from the two sides of the guillotine break are mixed together before entering the containment such that there is minimal steam released to the containment environment and the temperature of the water added to the containment sump is maximized. Therefore, from this description, the mass and energy releases for a similar type of break are manipulated to cover the potential uncertainties related to the break configuration and how this influences the specific attributes that must be evaluated to ensure that the containment is capable of remaining within its design basis envelope for all of the accident conditions considered.

As is evident from this brief discussion on the differences between the accident sequences, it is clear that the mass and energy releases must address the short term discharges for those accident conditions which give the peak pressure transient in the containment, as well as those mass and energy releases over the very long term that represent the containment response during cooldown of the RCS and decay heat removal from the containment. Those design basis mass and energy release evaluations related to the early discharge of steam and water to the containment atmosphere are a straightforward application of the design basis models which have been approved by the NRC. More specifically, these mass and energy releases represent sources (input) to the containment analysis with no feedback required from the containment part of the evaluation. However, for those accident sequences which result in the long term response of the containment, particularly after containment recirculation, the mass and energy releases from the RCS are dependent on the temperature of the containment sump, since this water is eventually injected into the RCS, even though it may pass through a heat exchanger prior to this injection. Since the sump temperature changes with time, long term containment response evaluations need to represent this temporal behavior. Specifically, the mass and energy releases need the sump water temperature history such that long term analyses properly incorporate the decreasing containment sump temperature.

The long-term (greater than 3600 seconds) mass and energy release calculations are performed through user defined input functions which are an option in the MAAP-DBA code. These input functions are used to incorporate the sump water cooling in the long term and are consistent with the Westinghouse methodology previously approved by the NRC. User defined functions are formulated to describe the long term discharge from the break for (a) a mixed discharge and (b) for an unmixed discharge of steam and water. In both cases, the flow rates used are those calculated by the design basis code and only the specific enthalpies of the discharge flows are calculated to represent the influence of the time dependent RCS injection temperature as the sump water cools. The user defined functions for the outlet specific enthalpies are as follows:

Unmixed: $h_{out,1} = (Q_{decay} + Q_{sg,1}) / W_1 + h_{in}$
$h_{out,2} = (Q_{RCS} + Q_{sg,23}) / W_2 + h_{in}$

Mixed: $h_{out} = (Q_{decay} + Q_{sg,1} + Q_{sg,23} + Q_{RCS}) / (W_1 + W_2) + h_{in}$

In these expressions, Q_{decay} is the core decay heat (Table 3.1-12), Q_{RCS} is the energy release rate from the RCS as sump water cools below 212°F, $Q_{sg,1}$ is the release rate of stored energy for the steam generator in the affected loop (from the upper head internals and upper elliptical head), $Q_{sg,23}$ is the release rate of remaining energy from the generators in the intact loops, W_1 is the steam flow rate through the affected steam generator calculated by the DBA LOCA code, W_2 is the total RCS injection flow rate minus W_1, and h_{in} is the water specific enthalpy for the recirculation flows entering the RCS.

For either of these break configurations, the long term evaluation requires that the outlet specific enthalpies be used as input. The specific enthalpy for the containment sump water is modified (decreased) if it passes through a heat exchanger before being injected into the RCS. The analysis for the long term evaluation requires this feedback from the containment evaluation to ensure that the mass and energy releases are representative of the accident conditions being evaluated.

Of course, the evaluations for the mass and energy discharges into the containment atmosphere are appropriately skewed towards the conservative (overstated) values of the mass and energy discharge rates. To ensure that this philosophy is preserved in the long term, the decay heat representation used in the above evaluations should be that which has the greatest decay heat coverage value for times greater than one hour after shutdown. Therefore, the 1979 ANS standard on decay heat power was used for all of the containment assessments as discussed in Chapter 3. Other than this, considerations related to whether the break discharge from the two sides of the postulated rupture are mixed or unmixed streams characterize the uncertainties to be evaluated for the mass and energy releases during the long term evaluations.

4.1.3 Influence of Varying Containment Operating Conditions

Another aspect of the evaluation is the spectrum of operating conditions that could be experienced by the containment at the time that the accident is initiated. For example, the containment pressure may vary between 12.8 and 14.2 psia. Furthermore, the initial containment atmosphere temperature could potentially be at either the maximum or minimum value. These types of operating parameters have an influence on the magnitude of the specific attribute being evaluated, and the different boundaries of these operating conditions need to be investigated to determine those conditions which maximize the challenge to the attribute being evaluated. One means of organizing this information is to develop a parameter matrix defining the conditions to be evaluated for a specific attribute such as peak pressure, maximum containment temperature transient, minimum available NPSH, etc. An example of such a parameter matrix is given in Table 4-3. This matrix also identifies those issues related to uncertainties in the paint thickness on steel and concrete heat sinks, the effective thermal resistance (the inverse of the effective heat transfer coefficient) for the paint on the steel and concrete heat sinks, uncertainties in the spray actuation setpoint, etc. Through such a parameter matrix, those conditions that maximize the challenge for a specific attribute can be defined and evaluated.

It is important to note that some of these evaluations may be insensitive to one or more of these parameters and in fact the specific limit that would be maximized for a given attribute may be somewhat sequence sensitive, i.e., different for a large LOCA vs. a MSLB. Therefore, the focus of such a parameter matrix is to identify those aspects of the containment conditions which have a major influence on the evaluation, with the others being identified as having essentially no influence.

4.2 MAAP-DBA MODELING APPROACH

As noted previously, the MAAP-DBA model utilizes both single node and multi-node models to address the spectrum of DBA analyses. Unit specific single node models are used for the peak pressure and peak temperature analyses and the unit specific multi-node models are used for those analyses which are influenced by water holdup in subcompartments. These two approaches are outlined below.

4.2.1 Single Node Containment Model

The design basis containment response calculations are implemented consistent with the intent of the Standard Review Plan. The containment peak pressure and temperature responses for large LOCA and main steamline breaks use the Tagami (Reference 1) and Uchida (Reference 2) heat transfer correlations to conservatively quantify the participation of the passive heat sinks. Implementation of these heat transfer correlations leads to the use of a single node containment model. Thus, the total containment volume and passive containment heat sinks are incorporated in a single node containment model that is applied for quantifying the peak pressure, peak gas temperature, and maximum containment liner temperature for the spectrum of main steamline break and large LOCA breaks. Furthermore, the containment liner temperature response is biased toward maximizing the energy transfer by using a multiplier of four (consistent with NUREG-0588) on the Tagami and Uchida heat transfer coefficients per the BVPS-1 and -2 current licensing basis.

The containment spray system impact varies by the accident sequence type. For the short-term large LOCA peak pressure and temperature responses, the time interval to the peak values is short and no containment spray system operation occurs prior to the peak condition. Conversely, main steamline break events, the peak conditions typically occur several hundred seconds following the postulated break. Thus, the quench spray system is activated and delivering spray to the containment atmosphere for most of the main steamline break sequence. A spray water droplet size of 1000 microns is modeled for heat and mass transfer. As a conservatism, no credit is taken for the recirculation sprays in determining the long-term temperature response for main steamline break calculations. For the main steamline breaks, the Uchida heat transfer correlation is used to describe the energy transfer rate to the passive containment heat sinks. The model used for the single node representation uses an 8% value (consistent with NUREG-0588 and NRC approved GOTHIC version 6.0a) for the percent of condensate that is revaporized into the gas space when superheated conditions are experienced.

The model credits the impact of airborne water droplets originating in the blowdown jet following LOCA accidents. During and following the blowdown transients, airborne water acts as a heat sink for steam to minimize or eliminate superheat as well as to condense some of the steam mass. With the large surface-to-volume ratio, and high water density and specific heat compared to steam, a relatively small droplet water mass is sufficient to eliminate steam superheat from the containment atmosphere. During the blowdown the fraction of available liquid airborne is specified to be 10% of the non-flashed liquid blowdown. It is noted that the MAAP-DBA code using Tagami heat transfer correlation demonstrates a low sensitivity using this value. A droplet size for the airborne liquid mass of 100 microns is assigned. Following the end of the blowdown, the fraction of available liquid airborne is specified to be 0%, which results in the remaining airborne water droplets mass decaying.

4.2.2 Multiple Node Containment Model

The assessment of some of the long-term containment response attributes is conducted with a multiple node containment model. Specifically, the large break LOCA NPSH, the small break LOCA NPSH, and the large and small break LOCA sump water temperature attributes implement a multiple node model. The sump water level and temperature histories are a key results to quantifying these specific attributes. Thus, the relative delivery rate and removal of water inventory from the containment sump and lower compartment influence the NPSH and sump temperature histories. Water hold-up from the break or spray injection sources in containment subcompartments directly influences the sump water level and temperature histories. Additionally, the distribution of containment sprays as they are collected on the operating deck floor can also influence these attributes. Thus, a multiple node containment configuration that identifies the elevations and sizes of junctions connecting the various containment regions is implemented for these evaluations.

The methods used to quantify the available NPSH for BVPS-1 and -2 are consistent with the current licensing basis. The BVPS-1 available NPSH method continues to include the overpressure credit, while the BVPS-2 available NPSH assessment does not credit overpressure.

The multiple node model uses natural convection heat transfer models for calculating the energy transfer rate to the containment heat sinks distributed through these multiple nodes. The natural convection heat transfer models are biased to minimize the calculated available NPSH. Like the single node model, the airborne fraction of LOCA blowdown is set to 10% with the droplet size set to 100 microns. Likewise, the spray system behavior is modeled in the same fashion as for the single node model.

The mass and energy releases from the primary system to containment for the spectrum of small and intermediate size LOCAs are generated using the MAAP-DBA code. The MAAP-DBA mass and energy release histories have been benchmarked with Westinghouse Small Break LOCA ECCS Evaluation Model (NOTRUMP). The mass and energy releases for the large break LOCA NPSH calculations are biased to yield the maximum sump water temperature by mixing the streams from the two sides of the guillotine break. As with the single node analyses, the mass and energy release histories for each side of the break are quantified by applying the NRC approved Westinghouse methodologies.

4.3 INPUT PARAMETERS AND ASSUMPTIONS

The input parameters for analysis are controlled through the combination of the run matrix and the parameter matrix. Tables 4-1 and 4-2 illustrates the run matrices that have been developed for BVPS-1 and -2 respectively. Table 4-3 shows the parameter matrix used for each of the types of analyses performed that controls the variations of the plant parameters (including the containment heat removal systems actuation signals and startup delay time). These attributes are combined with the specific objectives of a given analysis to ensure that the appropriate input conditions have been produced to provide the most challenging situation for the given analysis. As part of this evaluation, the mass and energy releases are also modified to provide the most challenging set of conditions. For example, those analyses which relate to the maximum sump temperature and/or the minimum NPSH require that the mass and energy releases be associated with a Double-Ended Pump Suction (DEPS) break, with the flow rates being assumed to be discharged from both sides of the double-ended break, and the respective enthalpies determined from the Westinghouse design basis analysis. Furthermore, the maximum enthalpy to the sump is established when the discharge from the two sides of the break are added together. This is accomplished in the input files through an include file which designates the mass and energy releases are mixed. This enables the information associated with the mass and energy releases to be clearly defined, since it modifies the mass and energy releases as supplied.

In the current BVPS-2 sub-atmospheric containment MSLB DBA analyses for peak pressure and temperature, the recirculation sprays are not credited. Similarly, in this evaluation for the BVPS-1 and -2 responses as atmospheric containments, these spray systems are also not credited in either the BVPS-1 or BVPS-2 MSLB evaluations (peak pressure and peak temperature). This is a conservatism in the modeling of these sequences.

4.4 PARAMETER FILES FOR THE BVPS UNITS

One principal input element for the MAAP-DBA code is the parameter file which defines the containment geometry, nominal operating conditions, pump curves, etc. The MAAP-DBA containment model is being supplied with large LOCA and MSLB mass and energy releases generated by Westinghouse codes which were previously approved by the NRC.

Prior to the development of the BVPS-specific modeling, detailed drawing and equipment specification reviews and site visits were conducted. The specific configurations of the BVPS-1 and -2 have been documented in unit-specific parameter files. These have been assembled, reviewed, and documented in a report prepared in compliance with 10CFR50, Appendix B requirements.

Based on detailed drawing reviews and site visits, consideration of the plant-specific features of both BVPS units and the current licensing basis, it was determined that each containment would most appropriately be represented with a single node model for assessing peak pressure and peak temperature and a multi-node model for those evaluations where water holdup is important. For the latter analyses an 18 node model was selected for BVPS-1 and a 17 node model for BVPS-2 (see Figures 4-1, 4-2 and 4-3). The difference is in the configuration of the refueling pool. Using these multi-node schemes provides a representation of the individual compartments for each of the three Reactor Coolant System (RCS) loops, the recirculation sump region, the reactor cavity region, the annular region outside of the cooling loops, and three nodes above the operating deck. Likewise, the flow paths between the containment compartments are also included, as junctions, where the junction areas and loss coefficients are based on the plant dimensions and are summarized in Table 4-4. A modification that is scheduled to be implemented in both containments is a 12 inch diameter drainage port in the biological shield between the reactor cavity and the lower compartment. With this in place, the potential water holdup in the reactor cavity will be minimized and the sump inventory will be enhanced. Both units are analyzed with the drainage port (junction area) included.

Major parts of the parameter file include the individual nodal volumes that make up the total containment volume, the volume vs. height function of these nodes such that water accumulation can be properly evaluated, the structural and containment heat sinks within these individual nodes, the surface characterization of the heat sink in terms of whether the surface is painted, how it is painted (number of layers, as well as the thickness and the thermal conductivity of each layer), whether the heat sink is concrete, steel or steel lined concrete, etc. Furthermore, the setpoints for system actuation, pump curves for the different configuration that are addressed, heat exchanger capacities, etc., are also contained in the unit-specific parameter file. Tables 4-5 and 4-6 tabulate the containment node volumes, metal heat sink areas and masses, and concrete heat sink areas and thicknesses included in the BVPS-1 and -2 containment model. The heat sinks include structural steel, concrete liners, ventilation ducts and supports, pipes, pipe supports and restraints, and heavy equipment.

The refueling cavity (nodes 14 and 18 in BVPS-1 and node 14 in BVPS-2) can accumulate and hold up spray water during a LOCA. This water holdup is an important consideration for the calculation of the NPSH for the recirculation spray (RS) and low head safety injection (LHSI) pumps and the containment sump water temperature history. The normal configuration during operation is to have the refueling cavity drains closed but several hatch covers in each refueling seal ring around the reactor pressure vessel flange are removed. Thus, water falling or draining into this region must accumulate until it can overflow through the open hatches in the refueling seal ring. The water flows through the open hatches and accumulates in the reactor cavity (node 1) until it overflows through the opening between the reactor cavity and the lower compartment (node 2) where the ECCS recirculation sump is located. Due to the differences in the shape of the refueling cavity in each unit, the holdup volumes are different, i.e., for BVPS-1 this volume is 13,600 gallons and for BVPS-2 it is 25,900 gallons.

The characteristics of the containment spray systems (header elevations and flow rates) are also included in the parameter file. A quench spray (QS) system (there are two QS trains in each design) is actuated on a containment high-high pressure signal and after a start delay, supplies cold water from the RWST to the quench spray ring header in containment. For BVPS-1 only, a portion of the quench spray pump discharge flow is diverted to the recirculation spray pumps suctions and the balance is directed to the spray header.

After a start delay, a recirculation spray (RS) system (there are two spray systems per train and two trains for each unit) is actuated, which directs water from the containment recirculation sump, through a heat exchanger, and then to the recirculation spray ring header in containment. Containment heat removal is accomplished by the RS heat exchanger.

4.5 DESCRIPTION OF ANALYSES

As mentioned above, the design basis evaluations for a large dry containment include evaluations of:

- peak containment pressure,
- peak containment gas temperatures,
- long-term temperatures within the containment (EQ envelope),
- peak liner temperatures,
- available NPSH for those pumps that take suction from the containment recirculation sump, and
- maximum sump and recirculation spray heat exchanger cooling water outlet temperatures for assessing piping temperatures.

Each attribute has been evaluated for the limiting design basis events using the MAAP-DBA containment models and the BVPS specific parameters as discussed above. Mass and energy releases for large LOCAs and MSLBs generated by previously approved Westinghouse codes were provided as input to the BVPS-specific MAAP-DBA containment model. As noted above, the MAAP-DBA containment model treats the mass and energy releases in a manner that results in conservative predictions of each of the containment attributes.

As discussed above, a unit-specific run matrix was used to assure the analyses examined the limiting mass and energy release and single active failure for each of the attributes evaluated. These run matrices are given in Tables 4-1 and 4-2 for BVPS-1 and -2, respectively and identify the break size, the reactor power level, and the single active failure included in each sequence definition. Note that some of the MSLB cases indicate two active failures, i.e., one related to the mass and energy release and the other to the containment heat removal capability. Design basis requirements only require single active failures be considered per sequence. However, to minimize the number of runs in the matrices, some cases included two active failures when that combination did not produce results that exceeded any design limit. Thus, the run matrix defines the specific accident sequence to be evaluated and the containment response to be quantified. Included in each evaluation are the plant operating conditions and the specification of the pumping capabilities, heat exchanger capacities, etc., that have different values depending upon the objective of the calculation. Thus, while the accident sequence in the run matrix may be the same, the conditions included in the evaluation for different containment attributes may differ. Therefore, each evaluation identified in the run matrix requires a separate computer calculation. To ensure that the specific containment conditions are included in a given analysis, the run matrix is supplemented by a parameter matrix which defines the boundaries of the operating conditions and uncertainties to be used for a particular analysis, see Table 4-3.

For example, when investigating the peak containment pressure, the initial containment pressure should be the maximum value consistent with the proposed plant Technical Specifications for atmospheric containment (14.2 psia); conversely when evaluating the minimum NPSH available, this pressure should be taken as the minimum value (12.8 psia). The various limits listed in the table are used for the individual analyses that are being performed for the specific accident initiator.

An additional analysis confirmed that inadvertent actuation of containment spray during normal plant operation would not result in containment pressure less than the minimum design pressure of 8 psia. That analysis treated the containment atmosphere as an ideal gas with actuation of spray at the minimum technical specification allowed RWST temperature.

4.6 ACCEPTANCE CRITERIA

An acceptance criterion was developed for each of the types of analyses being performed. These are as follows:

- peak containment pressure less than 45 psig and the pressure is less than half the peak pressure within 24 hours,
- peak containment gas temperature within equipment qualified capability,
- peak liner temperature less than 280°F,
- NPSH greater than the head required for safe operation of the respective pumps used in the different units,
- maximum sump temperature for small break LOCA is less than the value (190°F for BVPS-1 and 212°F for BVPS-2) used in the small break LOCA peak clad temperature (PCT) analysis,
- inadvertent actuation of containment spray during normal plant operation shall not result in containment pressure less than the minimum design pressure of 8 psia.

For the containment peak pressure, the acceptance criterion is the same for both units for all of the evaluations. When assessing the peak containment temperature and its comparison to the equipment qualification (EQ) profile, the issue is more related to time at temperature and component aging than a specific peak value. See Section 7.11 for further details on the EQ evaluation. For the peak liner temperature, the criterion has previously been established in the design and construction of the individual units related to the stresses in the liner under blowdown conditions. This criterion was set at 280°F and remains the criterion for the maximum liner temperature in these analyses.

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

- BVPS-1, IRS and ORS pumps, 9.8 ft,
- BVPS-1, LHSI pump, 10.6 ft,
- BVPS-2, RS pump, 15.0 ft.

4.7 RESULTS AND ACCEPTABILITY

4.7.1 Pressure and Temperature Response for LOCA

Each of the BVPS Design Bases LOCA events were investigated including:

- Large LOCA – double ended break in the hot leg (DEHL),
- Large LOCA – double ended break in the cold leg (DECL),
- Large LOCA – double ended break in the pump suction pipe (DEPS).

The large LOCA mass and energy release evaluations (see Chapter 3) determined that the DEHL and DEPS breaks bounded the DECL. The controlling attribute of the different types of breaks is simply the mass and energy release rates. Since the DEHL and DEPS releases are shown to bound those from a DECL break, detailed analyses were not required for the DECL.

Tables 4-7 through 4-11 summarize the sequence of events for the LOCA sequences for BVPS-1 with Tables 4-12 through 4-15 summarizing the results for BVPS-2.

Large break loss-of-coolant accidents have been analyzed for different break locations as well as different characteristics of the break. Specifically, the run matrix includes a Double-Ended Hot Leg (DEHL) break condition and a Double-Ended Pump Suction (DEPS) break in terms of the mass and energy releases to the containment. Furthermore, since there are a number of different containment attributes to be evaluated, these mass and energy releases are considered as either independent blowdown streams from each side of the double-ended rupture or assuming that the two streams are thoroughly mixed before being discharged to the containment. Typically, the unmixed is the mass and energy release which results in the maximum containment building pressurization, whereas mixing of the two streams results in the maximum sump temperature history. Also, the DEHL RCS break location results in the maximum containment pressure increase, whereas the DEPS break results in the highest containment sump temperature.

Both the quench spray and recirculation spray behavior are included in the LOCA assessments while only the quench sprays are considered in the MSLB results. The plant’s spray header elevations, spray pump curves, and spray coverage are all modeled. The spray drop size used was 1,000 microns, which is consistent with spray nozzle manufacturer's data which states that the mean equivalent diameter of the spray droplets is less than 1,000 microns.

Tables 4-16 and 4-17 summarize the peak containment pressures for the large break LOCA cases identified in the run matrices for the two units. All the pressures are reported as psig and referenced to an atmospheric pressure of 14.3 psi. As discussed previously, the DEHL condition results in the maximum pressure increase and results in a pressurization which is less than the design basis value of 45 psig, i.e., both units satisfy the acceptance criteria for peak containment pressure and temperature during a DBA large break LOCA sequence. Figures 4-4 and 4-5 show the calculated containment pressure history in the containment, and the gaseous atmosphere temperature.

4.7.2 Medium and SBLOCA Considerations

A spectrum of medium and small break LOCAs was also evaluated to determine the response of the containment depressurization system and impact on long term core cooling. Specifically, the analyses conducted included breaks in both the cold leg and hot leg in sizes from 1 to 12 inches. In these evaluations the mass and energy releases to containment were quantified utilizing the MAAP-DBA computer code that has been benchmarked with the Westinghouse small break computer code, NOTRUMP (see Chapter 9). Moreover, the RCS input information is taken from the Westinghouse data base. The derivation of the mass and energy input was consistent with that recently conducted for another plant.

Results from these small and medium size breaks show that the sump water inventory at the time that the recirculation spray pumps start is sufficient to support filling of the spray lines for both units. The limiting NPSH is obtained from medium and small break LOCA's (see Table 4-23 and 4-24). Furthermore, the results for medium and small break LOCA's for the balance of the containment response attributes were all bounded by the LBLOCA results. Moreover, in both units, there is sufficient NPSH for operation of the recirculation spray pumps.

4.7.3 Pressure and Temperature Response for MSLB

Given a postulated main steam line break accident, the mass and energy release to the containment is dependent upon a number of important parameters including the size of the main steam line rupture, the initial power level in the reactor (this influences the water inventory in the steam generator that could be discharged during the accident), and the potential for the reactor to return to power as a result of the rapid cooldown caused by the rapid depressurization of the affected steam generator. These are captured in the run matrix by considering different single failures, break sizes and initial reactor power. In addition, these assessments use a design basis methodology to evaluate the potential return to power of the reactor core using conservative reactivity coefficients. Moreover, since BVPS-1 has a flow limiter in the steam lines outside of the steam generator and BVPS-2 has this flow limiter installed within the generators, there is a difference between the two units with respect to the size of the main steam line breaks that must be considered. Therefore, the run matrix also includes these unit-specific differences.

Tables 4-18 and 4-19 are the peak pressure results for BVPS-1 and -2, respectively. As illustrated, all of the design basis main steam line break conditions have a peak containment pressure that is less than the 45 psig structural design basis for the BVPS units. Consequently, all of these design basis evaluations meet the acceptance criterion for peak containment pressure.

It is noted here that the above evaluations for main steam line breaks were all performed assuming that the containment recirculation sprays are not available during the entire transient. This conservatism is implemented to be consistent with the previous licensing bases for BVPS-2 which did not credit recirculation sprays for any main steam line break condition.

4.7.4 Equipment Qualification Envelope

The assessment of the peak temperature history for both the LOCA and main steam line breaks was performed by comparing the gaseous atmosphere temperature in each sequence with the equipment qualification capabilities for each unit. These results are illustrated for those BVPS-1 and -2 accident sequences with the maximum temperature history in Figures 4-6 through 4-13. The influence of the containment temperature is discussed in detail in Section 7.11.

4.7.5 Peak Liner Temperature Response

Assessments of the peak liner temperature for the large break LOCA and main steam line break MSLB sequences were performed by increasing the Tagami heat transfer coefficients and the Uchida heat transfer coefficient by a factor of four. MSLB sequences produced higher peak containment gas temperatures than the LOCA sequences, however, the LOCA sequences produce a higher pressure. The results are summarized for the sequences which resulted in the maximum temperature history in each unit in Table 4-20 and Table 4-21. The liner temperature is below the acceptance criterion of 280°F for each accident sequence. It should be noted that the MAAP-DBA spray model does not credit spray directly contacting the containment walls. Thus, only the hot condensate film is in contact with the containment liner and no benefit from the direct impact of the cooled containment spray is credited in these evaluations, which is a conservatism in the calculation.

4.7.6 Subcompartment Pressurization Response

The subcompartment analyses address three subcompartments inside containment; the reactor cavity; the steam generator compartments; and the pressurizer cubicles. Section 3.3.1 discusses the evaluation that was performed to determine the effect of the Extended Power Uprate (EPU) on the short-term LOCA-related mass and energy releases that support subcompartment analyses. The evaluation determined that only the BVPS-1 short-term mass and energy releases for the spray line break are increased for the EPU. A similar evaluation was performed and determined that the current BVPS-2 UFSAR mass and energy releases are bounding for the EPU without the need to adjust for lower RCS temperatures or higher RCS pressures.

A multi-node model was developed for the upper pressurizer spray cubicle in order to predict the pressure response following a spray line double-ended rupture (DER). The computer program THREED (Reference 3) was used to calculate the pressure transient using the Westinghouse mass and energy release data. The computer program THREED considers two-phase, two-component (steam-water-air) flow through the vents and accounts for the fluid inertial effects. The critical flow correlation selected for each vent path is the homogeneous equilibrium model (HEM). The computer program THREED is the same program that was used to license BVPS-2. The evaluation determined that sufficient margin exists within the compartment structural design to accommodate the increase in releases due to the power uprate.

Converting the containment to atmospheric conditions will increase the initial pressure. The use of a minimum initial pressure in the subcompartment analysis is conservative. Sensitivity studies reported in NUREG/CR-1199 (Reference 4) have shown that peak ΔP is usually higher for a lower initial pressure of the subcompartment. Therefore, the net effect of increasing the initial pressure will be a reduction of the predicted maximum ΔP. The current analysis remains bounding.

4.7.7 Combustible Gas Concentration Control and Monitoring

Currently the BVPS Technical Specifications contain requirements for hydrogen recombiners and hydrogen analyzers. Changes to these Technical Specifications are not being identified in this Licensing Report because it is anticipated that they will have been removed as a result of LARs 316 and 189 for BVPS-1 and -2, respectively, prior to implementation of containment conversion. These LARs were transmitted to the NRC by FENOC letter L-04-012, dated January 28, 2004, and requested an approval date of August 31, 2004. Approval of these LARs will remove the hydrogen recombiner Technical Specification requirements and relocate the hydrogen analyzer Technical Specification requirements to the BVPS Licensing Requirements Manuals.

LARs 316 and 189 state the BVPS units are sub atmospheric containments. Since the NRC feasibility study done in support of rulemaking, found " that combustible gas generated from design basis accidents was not risk significant for any containment type….", (page 54125 of Federal Register) FENOC has determined that no changes beyond those proposed in LARs 316 and 189 are required for conversion to an atmospheric containment.

Since FENOC letter L-04-012 requested NRC approval by August 31, 2004, and FENOC does not plan to implement atmospheric containment conversion on either BVPS unit until after that date, it is expected that the NRC will have approved LARs 316 and 189 in advance of conversion to atmospheric containments.

4.7.8 Maximum Sump Temperature/Available NPSH

The results from the assessment of the maximum sump temperature for the limiting LOCA sequence in both units were used to quantify the ECCS and recirculation spray piping temperatures at different times during the LOCA sequence. The results are summarized in Table 4-22.

One of the piping assessments for DBA evaluations is the piping temperature change when switchover to containment recirculation occurs, i.e., containment cold leg recirculation. This relates to the temperature change of the ECCS low head and high head injection piping, which prior to recirculation has temperatures equal to the RWST value, and at the onset to recirculation, experiences water flow through the stainless steel piping with the initial temperature being that of the containment sump water. At this time, the sump water temperature is decreasing due to heat removal from the containment recirculation spray heat exchangers.

The maximum recirculation spray heat exchanger cooling water outlet temperatures for BVPS-1 were found to be 196°F for Train B (limiting since larger heat exchangers than for Train A) and for BVPS-2 it was found to be 175°F for both trains. These conditions were determined by maximizing the sump water temperature and the rate of energy transfer to the recirculation spray heat exchanger cooling water. The key parameters used to achieve these conditions are defined in Table 4-3.

The assessment of the available NPSH for the LOCA sequences was performed by comparing the calculated available NPSH to the required NPSH for each pump. For BVPS-1, the inside recirculation spray (IRS), outside recirculation spray (ORS), and Low Head Safety Injection (LHSI) pumps require NPSH values of 9.8 ft, and 10.6 ft, respectively. Figure 4-14 presents the available NPSH history after the pump starts for these three systems for BVPS-1 and demonstrates that the required NPSH is satisfied. Consistent with the BVPS-1 licensing basis, containment overpressure was credited in the calculation of available NPSH. For BVPS-2, only the recirculation spray (RS) system takes suction from the recirculation sump and its required minimum NPSH is 15. 0 ft. Figure 4-15 illustrates that this NPSH requirement for the BVPS-2 RS system is satisfied. Consistent with the BVPS-2 licensing basis, only the sump water level and no containment over pressure is used to quantify the RS pump NPSH. The NPSH results are summarized in Table 4-23 and Table 4-24.

4.7.9 Sensitivity Analyses

Each design basis evaluation necessitates that the sensitivities associated with the spectrum of potential plant conditions be evaluated in order to determine the most challenging set of conditions related to the specific design basis attributes. Many of these are straightforward. For example, evaluation of the peak containment pressure necessitates that the calculation should use the minimum containment open volume, the maximum initial pressure, etc. However, with the various attributes evaluated, there are cases where maximum containment volume or the minimum initial pressure should be used. Generally these have a small influence on the particular attribute being investigated, the specific variation which provides the most challenging condition is a function of the accident sequence being evaluated. To address the influence of these changes, sensitivity analyses were performed to assess which of the input parameters/possible operating conditions would result in the greatest challenge to the containment operation for the attribute being investigated. Some of these were found to be easily dealt with as bounding values such as the paint thicknesses on the containment carbon steel liner and the concrete wall. Both the minimum and maximum values were evaluated and it was found that, with the exception of the maximum liner temperature, the maximum value always provided the most limiting case and therefore was used. When the maximum liner temperature was evaluated, maximum thicknesses were used on all surfaces except the liner, which used a minimum value. Others were more difficult to address as bounding values and were addressed through attribute specific sensitivity analyses.

4.7.10 Summary of Containment MAAP-DBA Results

The results for the containment analyses for the atmospheric configuration have been compared to the applicable acceptance criteria for both BVPS-1 and 2. The results for each attribute were calculated using the applicable set of bounding parameters for the mass and energy releases per the approved Westinghouse methodologies. Therefore, the results are considered to be bounding. The peak calculated containment pressure increase due to a spectrum of design basis LOCA and MSLB accidents does not exceed 45 psig in all cases for both BVPS units. Also, the calculated pressure transients demonstrate that the containment pressure is reduced to below one-half of the peak pressure within 24 hours. The maximum containment liner temperatures occurred for MSLB accidents. The maximum liner temperature for the entire spectrum of MSLB accidents is found to be less than the 280°F acceptance criterion. The NPSH available for the containment spray and ECCS pumps that take suction from the containment sump for the spectrum of LOCA accidents are all found to exceed the minimum required acceptance criteria.

4.7.11 Results of Inadvertent Spray Analysis

Analysis demonstrated that inadvertent operation of containment spray at the minimum technical specification allowed temperature of the RWST resulted in containment pressure greater than the minimum design value of 8 psia.

Thus the minimum design value of 8 psia or greater was satisfied for both Unit 1 and Unit 2.

4.7.12 References

1. Slaughterbeck, D. C., 1970, "Review of Heat Transfer Coefficients for Condensing Steam in a Containment Building Following a Loss-of-Coolant Accident," Idaho Nuclear Corporation, National Reactor Testing Station, Idaho Falls, ID.

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

3. Meyer, S.P., 1982, "Subcompartment Transient Response Code THREED," NU-092 (Proprietary).

4. NRC, 1979, "Subcompartment Analysis Procedures," NUREG/CR-1199 LA-8169-MS.

5. Beaver Valley Power Station Pre-Application Report dated November 2003 submitted to the NRC by FENOC letter L-03-188 on November 24, 2003.

Table 4-1
MAAP-DBA Run Matrix for Containment Analysis Beaver Valley – BVPS-1

Break Description	Power Level, %	Single Failure	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	Available NPSH			Maximum Sump Water Temp[(1)]	Maximum Service Water Outlet Temp.
						IRS	ORS	LHSI		
Loss-Of-Coolant Accident										
1L-DEHL MIN SI (unmixed)	100.6	DG				X	X			
1L1-DEHL MIN SI (unmixed)	100.6	LHSI				X	X			
2L-DEHL MAX SI (unmixed)	100.6	None				X	X			
3L-DEHL MIN SI (mixed)	100.6	DG				X	X			
4L-DEHL MAX SI (mixed)	100.6	None				X	X			
4L1-DEHL-MAX SI (mixed)	100.6	QS				X	X			
6L-DEPS MIN SI	100.6	DG	X	X	X	X	X	X	X	X
7L-DEPS MAX SI	100.6	CIB	X	X	X	X	X	X	X	
8L-DEHL	100.6	None	X	X						

Table 4-1 (Continued)
MAAP-DBA Run Matrix for Containment Analysis Beaver Valley – BVPS-1

Break Description	Power Level, %	Single Failure	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	Available NPSH			Maximum Sump Water Temp[(1)]	Maximum Service Water Outlet Temp.
						IRS	ORS	LHSI		
Main Steam Line Break										
1M-4.6 ft^2 DER	100.6	MSCV	X	X						
2M-4.6 ft^2 DER	100.6	CIB/MFIV	X	X						
3M-1.4 ft^2 DER	100.6	MSCV	X	X	X					
4M-1.4 ft^2 DER	100.6	CIB/MFIV	X	X						
5M-0.725 ft^2 Split	100.6	MSCV	X	X						
6M-0.725 ft^2 Split	100.6	CIB/MFIV	X	X	X					
7M-4.6 ft^2 DER	70	MSCV	X	X						
8M-4.6 ft^2 DER	70	CIB/MFIV	X	X						
9M-1.4 ft^2 DER	70	MSCV	X	X	X					
10M-1.4 ft^2 DER	70	CIB/MFIV	X	X						
11M-0.710 ft^2 Split	70	MSCV	X	X						
12M-0.710 ft^2 Split	70	CIB/MFIV	X	X	X					
13M-4.6 ft^2 DER	30	MSCV	X	X						
14M-4.6 ft^2 DER	30	CIB/MFIV	X	X						
15M-1.4 ft^2 DER	30	MSCV	X	X	X					
16M-1.4 ft^2 DER	30	CIB/MFIV	X	X						
17M-0.424 ft^2 Split	30	MSCV	X	X						

Table 4-1 (Continued)
MAAP-DBA Run Matrix for Containment Analysis Beaver Valley – BVPS-1

Break Description	Power Level, %	Single Failure	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	Available NPSH			Maximum Sump Water Temp.[(1)]	Maximum Service Water Outlet Temp.
						IRS	ORS	LHSI		
18M-0.424 ft^2 Split	30	CIB/MFIV	X	X	X					
19M-4.6 ft^2 DER	0	MSCV	X	X						
20M-4.6 ft^2 DER	0	CIB/MFIV30	X	X						
21M-1.4 ft^2 DER	0	MSCV	X	X						
22M-1.4 ft^2 DER	0	MFIV30	X	X						
22M1-1.4 ft^2 DER	0	CIB	X	X	X					
23M-1.4 ft^2 DER	0	MSCV/ MFIV30	X	X						
24M-0.209 ft^2 Split	0	CIB/MFIV30	X	X	X					

Single Failures – Failed Equipment

CIB	One train each, quench spray system (QSS), recirculation spray system (RSS)
DG	One train each, SI, QSS, RSS
LHSI	One LHSI train
MSCV	Main steam line check valve
MFIV	Main feedwater isolation valve
MFIV30	Main feedwater isolation valve (isolation time limited by FW bypass closure time of 30 seconds)
QS	One train QSS
SW	One SW pump (cooling lost in one train of RSS coolers)
(1)	Maximum sump temperature at the time each ESF system is started or the ECCS injection point is changed to the hot leg is used to assess the thermal response of the associated piping.

Table 4-2
MAAP-DBA Run Matrix for Containment Analysis Beaver Valley – BVPS-2

Break Description	Power Level, %	Single Failure	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	Maximum Sump Temp.[1]/Available NPSH	Maximum Service Water Outlet Temp.
Loss-Of-Coolant Accident							
1L-DEPS MIN SI	100.6	DG	X	X	X	X	X
2L-DEPS MAX SI	100.6	CIB	X	X	X	X	X
2L1-DEPS MAX SI	100.6	SW				X	
2L2 DEPS MAX SI	100.6	RELAY				X	
3L-DEHL	100.6	[2]	X	X		X	
Main Steam Line Break							
1M-1.069 ft^2 DER	100.6	MSIV/CIB	X	X			
2M-1.069 ft^2 DER	100.6	MFIV	X	X	X		
3M-0.753 ft^2 Split	100.6	CIB	X	X	X		
4M-0.753 ft^2 Split	100.6	MSIV	X	X			
5M-0.753 ft^2 Split	100.6	MFIV	X	X			
6M-1.069 ft^2 DER	70	MSIV/CIB	X	X			
7M-1.069 ft^2 DER	70	MFIV	X	X	X		
8M-0.757 ft^2 Split	70	CIB	X	X	X		
9M-0.757 ft^2 Split	70	MSIV	X	X			
10M-0.757 ft^2 Split	70	MFIV	X	X			
11M-1.069 ft^2 DER	30	MSIV/CIB	X	X			
12M-1.069 ft^2 DER	30	MFIV	X	X	X		

Table 4-2 (Continued)
MAAP-DBA Run Matrix for Containment Analysis Beaver Valley – BVPS-2

Break Description	Power Level, %	Single Failure	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	Maximum Sump Temp.(1)/Available NPSH	Maximum Service Water Outlet Temp.
13M-0.756 ft^2 Split	30	CIB	X	X	X		
14M-0.756 ft^2 Split	30	MSIV	X	X			
15M-0.756 ft^2 Split	30	MFIV	X	X			
16M-1.069 ft^2 DER	0	MSIV/CIB	X	X	X		
17M-1.069 ft^2 DER	0	MFIV	X	X	X		
18M-0.608 ft^2 Split	0	CIB	X	X	X		
19M-0.608 ft^2 Split	0	MSIV	X	X			
20M-0.608 ft^2 Split	0	MFIV	X	X			

Single Failures – Failed Equipment

CIB	One train each, quench spray system (QSS), recirculation spray system (RSS)
DG	One train each, SI, QSS, RSS
MSIV	One main steam isolation valve
MFIV	One main feedwater isolation valve
SW	One SW pump (cooling lost in one train of RSS coolers)
RELAY	One train RSS
(1)	Maximum sump temperature at the time each ESF system is started or the ECCS injection point is changed to the hot leg is used to assess the thermal response of the associated piping.
(2)	Peak pressure and temperature occur before any active failure occurs. For the maximum sump temperature and NPSH attributes a CIB or a SW failure were both evaluated and the limiting results were reported.

Table 4-3
Containment Parameter Limits for Design Basis Analyses

Design Input Parameter	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	NPSH Limits	Maximum Sump Water Temperature	Max Service Water Outlet Temperature
Containment Configuration and Initial Conditions						
Containment volume	Min[a]	Min	Min	Max[a] – LHSI Min RS	Max	Max
Initial containment pressure	Max[b]	Min	Max	Min[c]	Max	Max
Initial containment temperature	Max[e]	Max	Max	Max	Max	Max
Initial containment relative humidity	Min[f]	Min	Max[g]	Max	Max	Max
Steel liner to concrete gap effective heat transfer coefficient	Min[h]	Min	Min	Min	Min	Min
Paint thickness on carbon steel heat sinks	Max[i]	Max	Min[6] (liner) Max (non-liner)	Max	Max	Max
Effective heat transfer coefficient for the paint on the carbon steel	Min[j]	Min	Max[d] (liner) Min (non-liner)	Min	Min	Min
Paint thickness on concrete heat sinks	Max[k]	Max	Max	Max	Max	Max
Effective heat transfer coefficient for the paint on the concrete heat sinks	Min[aa]	Min	Min	Min	Min	Min
Zinc thickness on carbon steel	Max[m]	Max	Max	Max	Max	Max
RWST temperature	Max[n]	Max	Max	Max	Max	Max

Table 4-3 (Continued)
Containment Parameter Limits for Design Basis Analyses

Design Input Parameter	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	NPSH Limits	Maximum Sump Water Temperature	Max Service Water Outlet Temperature
Engineering Safeguards Actuation						
Containment high-high quench spray setpoint	Max[o]	Max	Max	Min[o] – LHSI Max – RS	Max	Max
Containment high (SI actuation, FW isolation and CIA) safety analysis limit setpoint range[7]	Max[p]	Max	Max	Max	N/A	N/A
Containment intermediate high-high (steam line isolation) safety analysis limit setpoint[7]	Max[y]	Max	Max	Max	N/A	N/A
Start delay for quench spray	Max[bb]	Max	Max	Min[cc]–RS (BVPS-1) Max – RS (BVPS-2) Max - LHSI	Max	Max
Quench spray flow rate	Min[jj]	Min	Min	Min – LHSI Max–RS (BVPS-1) Min–RS (BVPS-2)	Min	Min
Start delay for recirculation spray	Max[dd]	Max	Max	Min – IRS/ORS Max – LHSI	Max	Max
Recirculation spray HX UA (BTU/hr/°F)	N/A[2]	Min[3]	N/A[4]	Min[q] – LHSI Max[ii] – RS	Min	Max
Recirculating spray flow rate	N/A[2]	Min[3]	N/A[4]	Min[r] – LHSI Max[ff] – RS	Min	N/A
Recirculation HX cooling water temperature	N/A[2]	Max[3]	N/A[4]	Min[z] – RS Max[s] – LHSI	Max	Max

Table 4-3 (Continued)
Containment Parameter Limits for Design Basis Analyses

Design Input Parameter	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	NPSH Limits	Maximum Sump Water Temperature	Max Service Water Outlet Temperature
Recirculation spray HX cooling water flow rate	N/A(2)	Min(3)	N/A(4)	Min(t) – LHSI Max(gg) – RS	Min	Min
Range of usable RWST volume prior to switchover	N/A(2)	Min(3)	N/A(4)	Min(u)	Min	Min
Nitrogen gas mass accumulation (acc. Gas volume/initial pressure/initial temperature)	N/A(5)	Min(hh) (Min/ Min/Max)	Min (Min/ Min/Max)	Min (Min/Min/Max)	Min (Min/ Min/Max)	Min (Min/ Min/Max)
MAAP-DBA Model Parameters						
Quench spray droplet diameter	Max(v)	Max	Max	Min(w)	Min	Min

Table 4-3 (Continued)
Containment Parameter Limits for Design Basis Analyses

Design Input Parameter	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	NPSH Limits	Maximum Sump Water Temperature	Max Service Water Outlet Temperature

(a) MIN/MAX values = 1,751,734 ft^3/1,792,916 ft^3 (BVPS-1) 1,750,867 ft^3/1,792,383 ft^3 (BVPS-2)
(b) 14.2 psia
(c) 12.8 psia
(d) 2526 (BVPS-1), 2000 (BVPS-2) BTU/hr/ft^2 /ºF
(e) 105°F
(f) 15%
(g) 100%
(h) 100 BTU/hr/ft^2/°F
(i) 0.0065 inches (BVPS-1) and 0.009 inches (BVPS-2)
(j) 462 BTU/hr/ft^2/°F (BVPS-1) and 266 BTU/hr /ft^2/ºF (BVPS-2)
(k) 0.005 inches (BVPS-1) and 0.0105 and 0.036 inches for the containment walls/ceilings and floors respectively. (BVPS-2)
(m) 0.00251 inches (BVPS-1), 0.00234 inches (BVPS-2)
(n) 65°F
(o) 26.8 psia (max); 24.0 psia (min)
(p) 22 psia (max); 18 psia (min)
(q) Determined by pump curve – heat exchanger combination
(r) Variable: determined by pump curve.
(s) 90°F (BVPS-1) and 89°F (BVPS-2)
(t) 3900 gpm (BVPS-1) and 5300 gpm (BVPS-2)
(u) 317,000 gals (BVPS-1) and 411,500 gals (BVPS-2)
(v) 1 mm
(w) 0.1 mm
(x) N/A
(y) 24 psia (max); 20 psia (min)
(z) 32ºF

Table 4-3 (Continued)
Containment Parameter Limits for Design Basis Analyses

Design Input Parameter	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	NPSH Limits	Maximum Sump Water Temperature	Max Service Water Outlet Temperature

(aa) 200 BTU/hr/ft^2/°F (BVPS-1) and 42/144 BTU/hr/ft^2/°F (BVPS-2) floor/walls and ceilings.

(bb) 81.5 seconds (BVPS-1) and 74.5 seconds (BVPS-2)

(cc) 57.4 seconds (BVPS-1) and 34.2 seconds (BVPS-2)

(dd) 227 seconds (BVPS-1) and 637 seconds (BVPS-2)

(ee) N/A

(ff) Variable: determined by pump curve.

(gg) 4500 gpm (BVPS-1) and 6000 gpm (BVPS-2)

(hh) Min accumulator tank volume: 1436 ft^3 (BVPS-1) and 1444 ft^3 (BVPS-2)

(ii) Determined by the combination of pumps and heat exchanger

(jj) Variable: Determined by pump curve.

(kk) Variable: Determined by pump curve.

(1) N/A

(2) Has no influence on the LOCA peak pressure since sprays start after the peak value and no influence on MSLB since RSS not credited for MSLB sequences.

(3) Has no influence on the short term peak temperature since sprays start after the peak value. The long-term temperature response used this parameter.

(4) Has no influence on the peak liner temperature.

(5) Has no influence on LOCA peak pressure which occurs before N_2 is released; no influence for MSLB sequences since they do not activate the accumulators.

(6) While a minimum value is used on the carbon steel concrete liners, a maximum value is used on all other (non-liner) carbon steel heat sinks.

(7) Not used by MAAP-DBA when containment response is quantified given Westinghouse supplied mass and energy releases are used.

Table 4-4
Beaver Valley MAAP-DBA Parameter File Summary of Junction Flow Areas

Junction Number	Upstream Node Number	Downstream Node Number	Junction Flow Area (ft^2)	Junction Loss Coefficient
1	1	4	3.14	.618
2	1	5	6.0	.583
3	1	7	6.0	.583
4	1	9	6.0	.583
5	2	4	417.0	.510
6	2	5	160.0	.540
7	2	6	33.5	.526
8	2	7	155.0	.518
9	2	8	976.0	1.0
10	2	9	140.6	.545
11	2	10	1166.0	.894
12	2	11	1166.0	.894
13	3	5	28.0	.546
14	13	12	903.2	1.0
15	4	5	28.0	.546
16	4	10	517.8	.511
17	5	6	56.0	.535
18	5	15	384.8	.590
19	6	7	56.0	.535
20	6	11	56.0	.535
21	6	15	64.2	.617
22	7	15	389.4	.584
23	8	11	800.0	.516
24	8	15	673.4	.894
25	9	10	56.0	.535
26	9	15	389.4	.584
27	11	10	493.3	1.0
28	10	12	1257.3	.894

Table 4-4 (Continued)
Beaver Valley MAAP-DBA Parameter File Summary of Junction Flow Areas

Junction Number	Upstream Node Number	Downstream Node Number	Junction Flow Area (ft^2)	Junction Loss Coefficient
29	11	13	1257.3	.894
30	12	15	792.3	.535
31	13	15	130.0	.522
32	17	16	7980.0	1.0
33	16	15	8120.0	1.0
34	16	13	1822.0	1.0
35	16	12	1822.0	1.0
36	15	14	926 (BVPS-1) 1105 (BVPS-2)	1.0
37(U1) 37(U2)	18 14	2 2	0.0 (Refueling canal drain path)	.756
38	14	1	23.38 (BVPS-1); 10.02 (BVPS-2)	.538
41 (U1)	14	18	.0491	.572
42 (U1) 41 (U2)	1	2	0.785	.474
43 (U1)	14	18	324	1.0
44 (U1)	15	18	210	1.0

Note: Junction 39 represents design basis leakage and 40 is the containment failure junction set to add when containment pressure exceeds a pre-set value.

Table 4-5
Beaver Valley MAAP-DBA Parameter File Summary of Containment Nominal Volumes and Metal Heat Sinks

		BVPS-1 Net Free Volume (ft^3)	BVPS-2 Net Free Volume (ft^3)	Metal Heat Sinks	
				Mass (lbm)	Surface Area (ft^2)
1	Reactor cavity	11826	11826	425,023	2,975
2	Lower compartment	198039	200063	655,459	60,526
3	Instrument room	30872	30872	1,094	216
4	RHR platform	31264	31264	17,446	431
5	Loop C compartment	52311	52311	264,057	8,744
6	PZR compartment	48637	48637	50,933	2,431
7	Loop B compartment	49141	49141	267,633	11,410
8	RV head laydown area	45542	45542	17,075	2,898
9	Loop A compartment	51429	51429	284,087	11,462
10	Lower annulus south (BVPS-1) or north (BVPS-2) half	85457	85457	299,948	41,642
11	Lower annulus north (BVPS-1) or south (BVPS-2) half	85663	85663	280,947	37,869
12	Upper annulus south (BVPS-1) or north (BVPS-2) half	80082	80082	148,581	24,295
13	Upper annulus north (BVPS-1) or south (BVPS-2) half	80294	80294	224,330	23,716
14	Refueling cavity	26668	36620	131,960	5,522
15	Upper compartment cylindrical section	347071	347071	481,486	15,339
16	Upper compartment lower dome region	413,523	413,523	583,731	34,062
17	Upper compartment upper dome region	108,635	108,635	0	0
18	Refueling cavity[(3)]	7572	0	0	0
	TOTAL	1,754,025	1,758,430	4,133,789	283,537

Notes:

1. Metal heat sinks do not include major equipment, such as steam generators or RCS loop piping. Realistic heat sink values without any uncertainty included.
2. The containment steel liner mass is included with concrete heat sinks, therefore the liner mass is not reflected in the metal heat sink summary.
3. Unit 1 uses two nodes for refueling cavity because of HS configuration.

Table 4-6
Beaver Valley MAAP-DBA Parameter File Summary of Containment Concrete Heat Sinks

Heat Sink#	Description	Total Thickness ft	One-Sided Area ft^2	No. Sides Inside Ctmt	Total Area ft^2
1	Shield wall to lower	4.50	1,524	2	3,049
2	Refuel cavity wall to loop B	4.00	958	2	1,915
3	Shield wall to loop C	4.50	195	2	389
4	Refuel cavity wall to PZR	4.00	924	2	1,848
5	Shield wall to loop B	4.50	211	2	421
6	Shield wall to RV laydown	4.50	62	2	124
7	Shield wall to loop A	4.50	155	2	309
8	Instrument tunnel to lower	3.00	954	2	1,908
9	Refuel cavity wall to loop C	4.00	826	2	1,653
10	Reactor cavity floor(1)	10.00	621	1	621
11	14 crane wall support columns	2.00	1,646	2	3,293
12	Lower compartment floor	10.00	10,094	1	10,094
13	Lower compartment outer wall(1)	4.50	8,445	1	8,445
14	Instrument room floor	4.00	1,017	2	2,034
15	Instrument room wall to loop C	3.25	929	2	1,858
16	Instrument room wall to loop A	3.25	704	2	1,408
17	Instrument room crane wall	2.00	1,545	2	3,091
18	Instrument room ceiling	2.00	923	2	1,846
19	Loop C floor	4.50	912	2	1,824
20	Loop C wall to PZR	3.00	1,043	2	2,085
21	Loop C crane wall	2.75	2,290	2	4,579
22	SG cubicle support columns	3.50	633	2	1,265
23	Loop C ceiling	2.00	923	2	1,846
24	PZR floor	2.00	880	2	1,761
25	PZR wall to loop B	3.00	1,253	2	2,506
26	PZR crane wall	2.00	2,568	2	5,136
27	PZR intermediate deck	4.00	984	2	1,969
28	PZR ceiling	2.00	923	2	1,846
29	Loop B floor	4.50	914	2	1,828
30	Loop B wall to RV head laydown	3.00	717	2	1,435

Table 4-6 (Continued)
Beaver Valley MAAP-DBA Parameter File Summary of Containment Concrete Heat Sinks

Heat Sink#	Description	Total Thickness ft	One-Sided Area ft^2	No. Sides Inside Ctmt	Total Area ft^2
31	Loop B crane wall	2.75	1,768	2	3,536
32	Loop B intermediate roof	6.00	131	2	261
33	Loop B ceiling	2.00	923	2	1,846
34	RV head laydown wall to fuel transfer canal	4.00	885	2	1,771
35	RV head laydown crane wall	2.75	915	2	1,829
36	RV head laydown ceiling	2.00	923	2	1,846
37	Loop A floor	4.50	1,082	2	2,165
38	Loop A crane wall	2.75	2,114	2	4,228
39	Loop A wall to fuel transfer canal	4.00	1,623	2	3,247
40	Loop A interior walls	2.00	271	2	543
41	Loop A ceiling	2.00	923	2	1,846
42	Lower annulus south half outer wall(1)	4.50	8,432	1	8,432
43	Lower annulus north half outer wall(1)	4.50	8,432	1	8,432
44	Upper annulus south half crane wall	2.75	8,226	2	16,452
45	Upper annulus south half outer wall(1)	4.50	7,569	1	7,569
46	Upper annulus north half crane wall	2.75	9,038	2	18,076
47	Upper annulus north half outer wall(1)	4.50	7,569	1	7,569
48	Fuel transfer canal floor	4.00	471	2	942
49	Lower dome outer wall	2.50	9,929	1	9,929
50	Upper dome outer wall	2.50	8,774	1	8,774
51	Pressurizer interior walls	2.00	527	2	1,054
52	Instrument room interior wall	1.25	147	2	295
53	RV laydown to Loop A misc wall	3.00	164	2	328
54	Support beam at 718'-6"	4.50	274	2	548
55	Cubicle walls above op. deck	1.50	3,262	2	6,523
56	RHR room wall to Loop C	3.25	539	2	1,078
57	RHR room wall to Loop A	3.25	568	2	1,135
58	RHR room crane wall	2.75	1,342	2	2,684

Table 4-6 (Continued)
Beaver Valley MAAP-DBA Parameter File Summary of Containment Concrete Heat Sinks

Heat Sink#	Description	Total Thickness ft	One-Sided Area ft^2	No. Sides Inside Ctmt	Total Area ft^2
59	Refuel cavity wall to upper annulus south half	4.00	1,033	2	2,066
60	Refuel cavity wall to upper annulus north half	4.00	1,181	2	2,361
61	Containment shell sections with embedment plates in lower compartment(1)	4.50	338	1	338
62	Containment shell sections with embedment plates in lower south annulus(1)	4.50	492	1	492
63	Containment shell sections with embedment plates in lower north annulus(1)	4.50	492	1	492
64	Containment shell sections with embedment plates in upper south annulus(1)	4.50	525	1	525
65	Containment shell sections with embedment plates in upper north annulus(1)	4.50	525	1	525
66	Lower dome sections with embedment plates	2.50	3,313	1	3,313
67	Upper dome sections with embedment plates	2.50	2,922	1	2,922
68	Wall Adjacent to Reactor Enclosure	3.00	705	2	1,410
69	Cubicle	1.50	1,590	2	3,179
70	Elevator Pit	1.00	94	2	188
71	Unlined portion of lower compartment outer wall	4.50	640	1	640
72	Unlined portion of lower annulus south half outer wall	4.50	836	1	836
73	Unlined portion of lower annulus north half outer wall	4.50	836	1	836
74	Unlined portion of upper annulus south half outer wall	4.50	856	1	856
75	Unlined portion of upper annulus north half outer wall	4.50	856	1	856
	TOTAL				217,155

Notes:

(1) Includes painted carbon steel liner and gap resistance between liner and concrete.

Table 4-7
BVPS-1 Double-Ended Hot-Leg Break Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.0	Low Pressurizer Pressure SI Setpoint is reached (1745 psia)
1.8	Containment High-High Setpoint is reached
11.9	Broken Loop Accumulator Begins Injecting Water
12.1	Intact Loop Accumulator Begins Injecting Water
15.7	Peak Containment Pressure During Blowdown
22.2	End of Blowdown Phase

Table 4-8
BVPS-1 Double-Ended Pump Suction Break Minimum Safeguards Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
1.8	Containment High-High Setpoint is reached
3.0	Low Pressurizer Pressure SI Setpoint is reached (1745 psia)
12.9	Broken Loop Accumulator Begins Injecting Water
13.3	Intact Loop Accumulator Begins Injecting Water
17.1	Peak Containment Pressure During Blowdown
21	End of Blowdown Phase
30.3	Safety Injection Begins
56.5	Accumulator Water Injection Ends
83.2	Quench Spray is initiated
225.8	End of Reflood Phase
229.9	Recirculation Spray is initiated
2900.0	ECCS Recirculation Begins
3600(1)	Transient Modeling Terminated

(1) Except for long term attributes such as EQ profiles, sump water, and temperature.

Note: This sequence of events is based on the LOCA M&E methodology described in WCAP 10325 (Section 3.1, Reference 2). This methodology was used to determine the recirculation switchover setpoints in the Technical Specifications.

Table 4-9
BVPS-1 Double-Ended Pump Suction Break Maximum Safeguards Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.0	Low Pressurizer Pressure SI Setpoint is reached (1745 psia)
1.8	Containment High-High Setpoint is reached
12.9	Broken Loop Accumulator Begins Injecting Water
13.3	Intact Loop Accumulator Begins Injecting Water
17.1	Peak Containment Pressure During Blowdown
21.0	End of Blowdown Phase
30.3	Safety Injection Begins
56.9	Accumulator Water Injection Ends
83.2	Quench Spray is initiated
221.3	End of Reflood Begins
229.9	Recirculation Spray is initiated
2500.0	ECCS Recirculation Begins
3600(1)	Transient Modeling Terminated

(1) Except for long term attributes such as EQ profiles, sump water, and temperature.

Note: This sequence of events is based on the LOCA M&E methodology described in WCAP 10325 (Section 3.1, Reference 2). This methodology was used to determine the recirculation switchover setpoints in the Technical Specifications.

Table 4-10
BVPS-1 Double-Ended Hot-Leg Break (Minimum Safeguards) Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.0	Low Pressurizer Pressure SI Setpoint is reached (1745 psia)
4.6	Containment High-High Setpoint is reached
11.9	Broken Loop Accumulator Begins Injecting Water
12.1	Intact Loop Accumulator Begins Injecting Water
17.4	Peak Containment Pressure During Blowdown
22.2	End of Blowdown Phase
22.4	Safety Injection begins
57.6	Accumulator Water Injection Ends
86.6	Quench Spray is initiated
157.2	End of Reflood
301.2	Recirculation Spray is initiated
1500	ECCS Recirculation Begins
6214.1	Quench Spray is terminated
1000000	Transient Modeling Terminated

Note: This sequence of events is based on the hot leg break mass and energy releases utilizing the LOCA M&E methodology described in WCAP 8264 (Section 3.1, Reference 5). The switchover to recirculation is assumed to be 1500 seconds.

Table 4-11
BVPS-1 Double-Ended Hot-Leg Break (Maximum Safeguards) Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.0	Low Pressurizer Pressure SI Setpoint is reached (1745 psia)
4.6	Containment High-High Setpoint is reached
11.9	Broken Loop Accumulator Begins Injecting Water
12.1	Intact Loop Accumulator Begins Injecting Water
17.4	Peak Containment Pressure During Blowdown
22.2	End of Blowdown Phase
22.4	Safety Injection Begins
57.6	Accumulator Water Injection Ends
86.6	Quench Spray is initiated
129.6	End of Reflood
301.2	Recirculation Spray is initiated
1500	ECCS Recirculation Begins
5715	Quench Spray is terminated
1000000	Transient Modeling Terminated

Note: This sequence of events is based on the hot leg break mass and energy releases utilizing the LOCA M&E methodology described in WCAP 8264 (Section 3.1, Reference 5). The switchover to recirculation is assumed to be 1500 seconds.

Table 4-12
BVPS-2 Double-Ended Hot-Leg Break Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.0	Low Pressurizer Pressure SI Setpoint is reached (1760 psia)
1.8	Containment High-High Setpoint is reached
11.5	Broken Loop Accumulator Begins Injecting Water
11.6	Intact Loop Accumulator Begins Injecting Water
18.2	Peak Containment Pressure During Blowdown
19.2	End of Blowdown Phase

Table 4-13
BVPS-2 Double-Ended Pump Suction Break Minimum Safeguards Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.0	Low Pressurizer Pressure SI Setpoint is reached (1760 psia)
1.7	Containment High-High Setpoint is reached
12.9	Broken Loop Accumulator Begins Injecting Water
13.0	Intact Loop Accumulator Begins Injecting Water
17.3	Peak Containment Pressure During Blowdown
21.4	End of Blowdown Phase
27.0	Safety Injection Begins
63.2	Accumulator Water Injection Ends
76.3	Quench Spray is initiated
214.2	End of Reflood Phase
642.5	Recirculation Spray is initiated
2948	ECCS Recirculation Begins
3600(1)	Transient Modeling Terminated

(1) Except for long term attributes such as EQ profiles, sump, and water temperature.

Note: This sequence of events is based on the LOCA M&E methodology described in WCAP 10325 (Section 3.1, Reference 2). This methodology was used to determine the recirculation switchover setpoints in the Technical Specifications.

Table 4-14
BVPS-2 Double-Ended Pump Suction Break Maximum Safeguards (CIB Failure) Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.0	Low Pressurizer Pressure SI Setpoint is reached (1760 psia)
1.7	Containment High-High Setpoint is reached
12.9	Broken Loop Accumulator Begins Injecting Water
13.0	Intact Loop Accumulator Begins Injecting Water
17.3	Peak Containment Pressure During Blowdown
21.4	End of Blowdown Phase
27.0	Safety Injection Begins
63.9	Accumulator Water Injection Ends
76.3	Quench Spray is initiated
213.4	End of Reflood Phase
642.7	Recirculation Spray is initiated
2782	ECCS Recirculation Begins
3600[(1)]	Transient Modeling Terminated

(1) Except for long term attributes such as EQ profiles, sump, and water temperature.

Note: This sequence of events is based on the LOCA M&E methodology described in WCAP 10325 (Section 3.1, Reference 2). This methodology was used to determine the recirculation switchover setpoints in the Technical Specifications.

Table 4-15
BVPS-2 Double-Ended Pump Suction Break Maximum Safeguards (SW Failure) Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.0	Low Pressurizer Pressure SI Setpoint is reached (1760 psia)
4.3	Containment High-High Setpoint is reached
12.9	Broken Loop Accumulator Begins Injecting Water
13.0	Intact Loop Accumulator Begins Injection Water
17.9	Peak Containment Pressure During Blowdown
21.4	End of Blowdown Phase
27.0	Safety Injection Begins
63.9	Accumulator Water Injection Ends
87.4	Quench Spray is initiated
213.4	End of Reflood
721.3	Recirculation Spray is initiated
2128	ECCS Recirculation Begins
3600(1)	Transient Modeling Terminated

(1) Except for long term attributes such as EQ profiles, sump, and water temperature.

Note: This sequence of events is based on the LOCA M&E methodology described in WCAP 10325 (Section 3.1, Reference 2). This methodology was used to determine the recirculation switchover setpoints in the Technical Specifications.

Table 4-16
MAAP-DBA Containment Peak Pressure Results for a Design Basis Large Break LOCA
Beaver Valley – BVPS-1

Description	Power Level, %	Single Failure	Peak Pressure (psig) (1)
6L-DEPS MIN SI	100.6	DG	42.0
7L-DEPS MAX SI	100.6	CIB	42.0
8L-DEHL	100.6	None	43.3

Single Failures – Failed Equipment

CIB One train each, QSS, RSS
DG One train each, SI, QSS, RSS
(1) Gauge pressure is referenced to 14.3 psi atmospheric pressure.

Table 4-17
MAAP-DBA Containment Peak Pressure Results for a Design Basis Large Break LOCA
Beaver Valley – BVPS-2

Description	Power Level, %	Single Failure	Peak Pressure (psig)$^{(1)}$
1L-DEPS MIN SI	100.6	DG	42.4
2L-DEPS MAX SI	100.6	CIB	42.4
3L-DEHL	100.6	None	44.9

Single Failures – Failed Equipment

CIB One train each, QSS, RSS

DG One train each, SI, QSS, RSS

(1) Gauge pressure is referenced to 14.3 psi atmospheric pressure.

Table 4-18
MAAP-DBA Peak Pressure Results for a Design Basis Main Steam Line Break
Beaver Valley – BVPS-1

Description	Power Level, %	Single Failure*	Peak Pressure (psig)
1M-4.6 ft^2 DER	100.6	MSCV	37.8
2M-4.6 ft^2 DER	100.6	CIB/MFIV	31.3
3M-1.4 ft^2 DER	100.6	MSCV	40.5
4M-1.4 ft^2 DER	100.6	CIB/MFIV	33.2
5M-0.725 ft^2 Split	100.6	MSCV	33.8
6M-0.725 ft^2 Split	100.6	CIB/MFIV	32.2
7M-4.6 ft^2 DER	70	MSCV	38.7
8M-4.6 ft^2 DER	70	CIB/MFIV	32.5
9M-1.4 ft^2 DER	70	MSCV	40.9
10M-1.4 ft^2 DER	70	CIB/MFIV	33.8
11M-0.710 ft^2 Split	70	MSCV	36.3
12M-0.710 ft^2 Split	70	CIB/MFIV	34.8
13M-4.6 ft^2 DER	30	MSCV	39.0
14M-4.6 ft^2 DER	30	CIB/MFIV	32.9
15M-1.4 ft^2 DER	30	MSCV	42.6
16M-1.4 ft^2 DER	30	CIB/MFIV	35.1
17M-0.424 ft^2 Split	30	MSCV	33.6
18M-0.424 ft^2 Split	30	CIB/MFIV	34.2
19M-4.6 ft^2 DER	0	MSCV	36.2
20M-4.6 ft^2 DER	0	CIB/MFIV30	32.1
21M-1.4 ft^2 DER	0	MSCV	41.3
22M-1.4 ft^2 DER	0	MFIV30	33.8
22M1-1.4 ft^2 DER	0	CIB	33.8
23M-0.209 ft^2 Split	0	MSCV/MFIV30	26.1
24M-0.209 ft^2 Split	0	CIB/MFIV30	30.3

Single Failures – Failed Equipment

CIB	One train QSS (RSS is conservatively not credited for MSLB evaluations)
DG	One train each, SI, QSS
MSCV	Main steam line check valve
MFIV	One main feedwater isolation valve
MFIV30	Main feedwater isolation valve (isolation time limited by FW bypass closure time of 30 seconds)
*	Some of these cases assumed two active failures, one for M&E release and the other for containment response. This is a conservatism that helps control the number of cases in the run matrix. The M&Es for the double failure cases were not significantly different than if only a single failure had been assumed.

Table 4-19
MAAP-DBA Peak Pressure Results for a Design Basis Main Steam Line Break
Beaver Valley – BVPS-2

Description	Power Level, %	Single Failure*	Peak Pressure (psig)
1M-1.069 ft^2 DER	100.6	MSIV	36.8
2M-1.069 ft^2 DER	100.6	MFIV/CIB	36.6
3M-0.753 ft^2 Split	100.6	CIB	31.5
4M-0.753 ft^2 Split	100.6	MSIV	32.3
5M-0.753 ft^2 Split	100.6	MFIV	31.5
6M-1.069 ft^2 DER	70	MSIV	37.0
7M-1.069 ft^2 DER	70	MFIV/CIB	36.8
8M-0.757 ft^2 Split	70	CIB	32.8
9M-0.757 ft^2 Split	70	MSIV	33.6
10M-0.757 ft^2 Split	70	MFIV	32.4
11M-1.069 ft^2 DER	30	MSIV	39.3
12M-1.069 ft^2 DER	30	MFIV/CIB	38.6
13M-0.756 ft^2 Split	30	CIB	35.4
14M-0.756 ft^2 Split	30	MSIV	36.3
15M-0.756 ft^2 Split	30	MFIV	34.6
16M-1.069 ft^2 DER	0	MSIV	37.9
17M-1.069 ft^2 DER	0	MFIV/CIB	36.7
18M-0.608 ft^2 Split	0	CIB	33.3
19M-0.608 ft^2 Split	0	MSIV	31.7
20M-0.608 ft^2 Split	0	MFIV	31.0

Single Failures – Failed Equipment

CIB	One train QSS
DG	One train each, SI, QSS, SW
MSIV	One main steam isolation valve
MFIV	One main feedwater isolation valve
*	Some of these cases assumed two active failures, one for M&E release and the other for containment response. This is a conservatism that helps control the number of cases in the run matrix. The M&Es for the double failure cases were not significantly different than if only a single failure had been assumed.

Table 4-20
MAAP-DBA Peak Liner Temperatures for BVPS-1

LOCA		MSLB	
Sequence	**Peak Liner Temperature (°F)**	**Sequence**	**Peak Liner Temperature (°F)**
Case6L_Tagami_MLT	235.7	Case3M_Uchida_MLT_NRS	246.5
Case7L_Tagami_MLT	235.8	Case6M_Uchida_MLT_NRS	234.4
		Case9M_Uchida_MLT_NRS	249.7
		Case12M_Uchida_MLT_NRS	241.9
		Case15M_Uchida_MLT_NRS	254.1
		Case18M_Uchida_MLT_NRS	242.6
		Case22M1_Uchida_MLT_NRS	237.6
		Case24M_Uchida_MLT_NRS	239.7

Table 4-21
MAAP-DBA Peak Liner Temperatures for BVPS-2

LOCA		MSLB	
Sequence	**Peak Liner Temperature (°F)**	**Sequence**	**Peak Liner Temperature (°F)**
Case1L_Tagami_MLT	235.7	Case2M_Uchida_MLT_NRS	240.4
Case2L_Tagami_MLT	235.7	Case3M_Uchida_MLT_NRS	232.0
		Case7M_Uchida_MLT_NRS	242.6
		Case8M_Uchida_MLT_NRS	236.1
		Case12M_Uchida_MLT_NRS	246.0
		Case13M_Uchida_MLT_NRS	241.3
		Case16M_Uchida_MLT_NRS	247.7
		Case17M_Uchida_MLT_NRS	245.5
		Case18M_Uchida_MLT_NRS	243.2

Table 4-22
MAAP-DBA ECCS and Recirculation Spray Piping Temperatures for Beaver Valley

Case	Prior to Recirculation Spray System Start	Prior to Cold Leg Recirculation Transition	After Cold Leg Recirculation Transition	After Hot Leg Recirculation Transition(1)
BVPS-1				
6L	250°F	244°F	183.5°F	146°F
7L	250°F	245°F	178.6°F	147°F
BVPS-2				
1L	248°F	246°F	161.3°F	126°F
2L	248°F	244°F	160.4°F	127°F
2L1	248°F	242°F	168.6°F	132°F
2L2	248°F	242°F	162.9°F	128°F
3L	244°F	239°F	169.3°F	113°F

Note:
(1) Switch to containment hot leg recirculation is assumed to be six (6) hours from accident initiation.

Table 4-23
MAAP-DBA Minimum Recirculation Spray and LHSI Pump NPSH Values for a Design Basis Large Break LOCA Beaver Valley – BVPS-1

Description	Power Level, %	Single Failure	IRS NPSH (ft)	ORS NPSH (ft)	LHSI NPSH (ft)
1L-DEHL MIN SI (1)	100.6	DG	20.4	14.6	---
1L1-DEHL MIN SI (1)	100.6	LHSI	21.9	16.1	---
2L-DEHL MAX SI (1)	100.6	None	19.7	13.9	---
3L-DEHL MIN SI (1)	100.6	DG	19.5	13.6	---
4L-DEHL MAX SI (1)	100.6	None	20.1	14.3	---
4L1-DEHL MAX SI (1)	100.6	QS	22.7/15.1(2)	16.8/9.3(2)	---
6L-DEPS MIN SI	100.6	DG	21.4	15.5	27.17
7L-DEPS MAX SI	100.6	CIB	21.3	15.4	29.36
RS_12IN_QS_IRS_HL(3)	100.6	QS	18.1	---	---
RS_12IN_QS_HL(3)	100.6	QS	---	12.3	---

Single Failures – Failed Equipment

CIB	One train each, QSS, RSS
DG	One train each, SI, QSS, RSS
LHSI	One LHSI train
QS	One Train of QS

Note:

(1) The M&Es provided for these cases are very conservative. The M&Es used an injection water temperature (supplied from the RWST) of 105°F and included the stored energy in the steam generator upper head metal during the first hour of the M&E releases.

(2) A QS failure results in the loss of enhancement flow to one train of RS. The values shown reflect the results for the train with enhancement flow and the train without enhancement flow. The minimum NPSH for the ORS pump without enhancement flow drops below the minimum required of 9.8 feet for a brief period. While no failure is expected to result from this, this case is bounded by minimum safeguards assumptions since 3 of 4 RS pumps meet the criteria.

(3) Limiting medium and small break LOCA sequence.

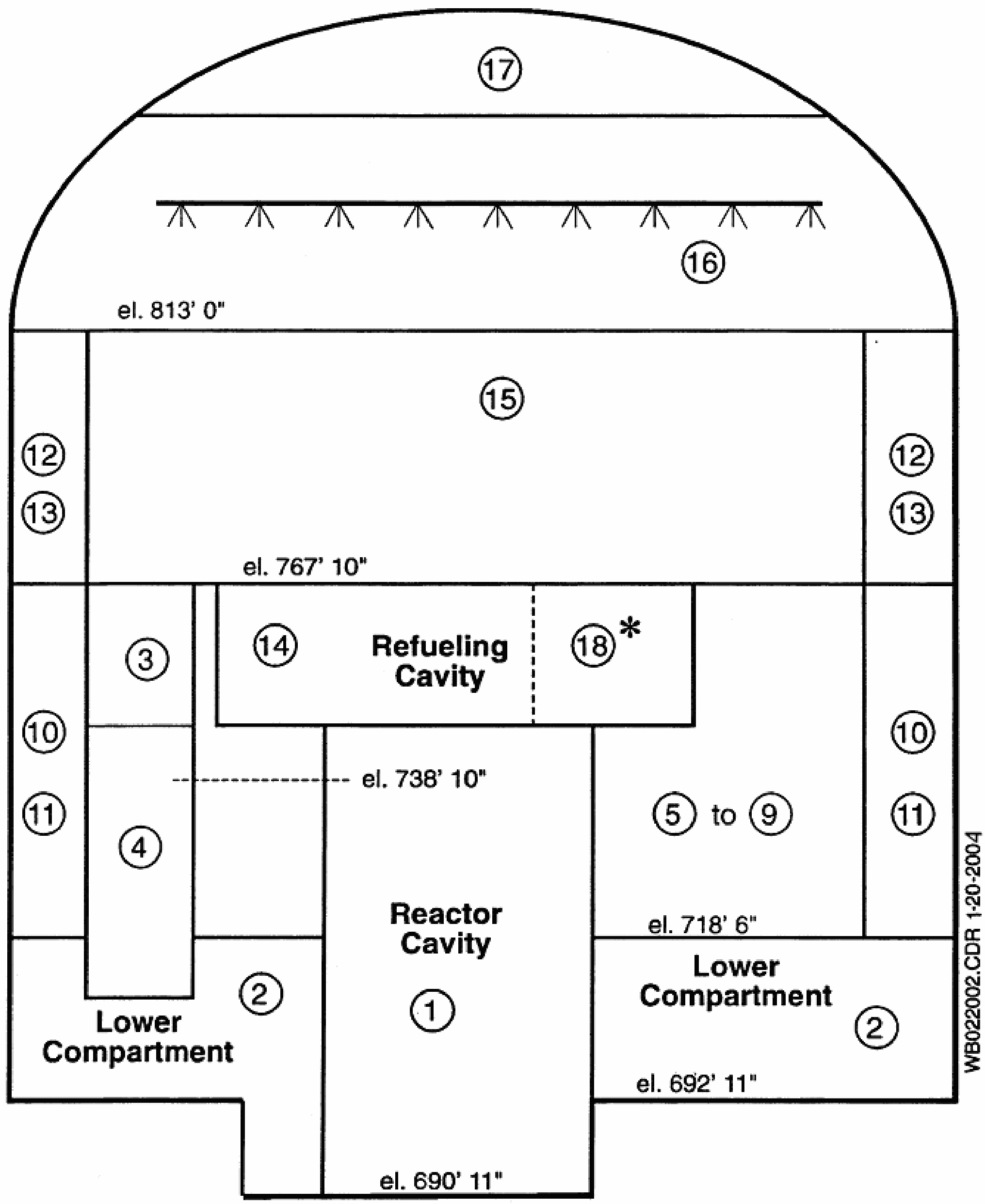
Table 4-24
MAAP-DBA Minimum Recirculation Spray NPSH for a Design Basis Large Break LOCA
Beaver Valley – BVPS-2

Description	Power Level, %	Single Failure	RS NPSH (ft)
1L-DEPS MIN SI	100.6	DG	15.8
2L-DEPS MAX SI	100.6	CIB	16.1
2L1-DEPS MAX SI	100.6	SW	19.4
2L2-DEPS MAX SI	100.6	RELAY	16.2
3L-DEHL (break into node #5)	100.6	None	19.3
RS_03IN_DG_HL(1)	100.6	DG	15.1

Single Failures – Failed Equipment

CIB	One train each, QSS, RSS
DG	One train each, SI, QSS, RSS
SW	One train of service water (SW)
RELAY	One train of RSS due to start relay failure

(1) Limiting medium and small break LOCA sequence.



*Refueling cavity is modeled as two nodes (14 and 18) in Unit 1 and a single node (14) in Unit 2.

Figure 4-1
MAAP-DBA Containment Nodalization for BVPS-1 and -2

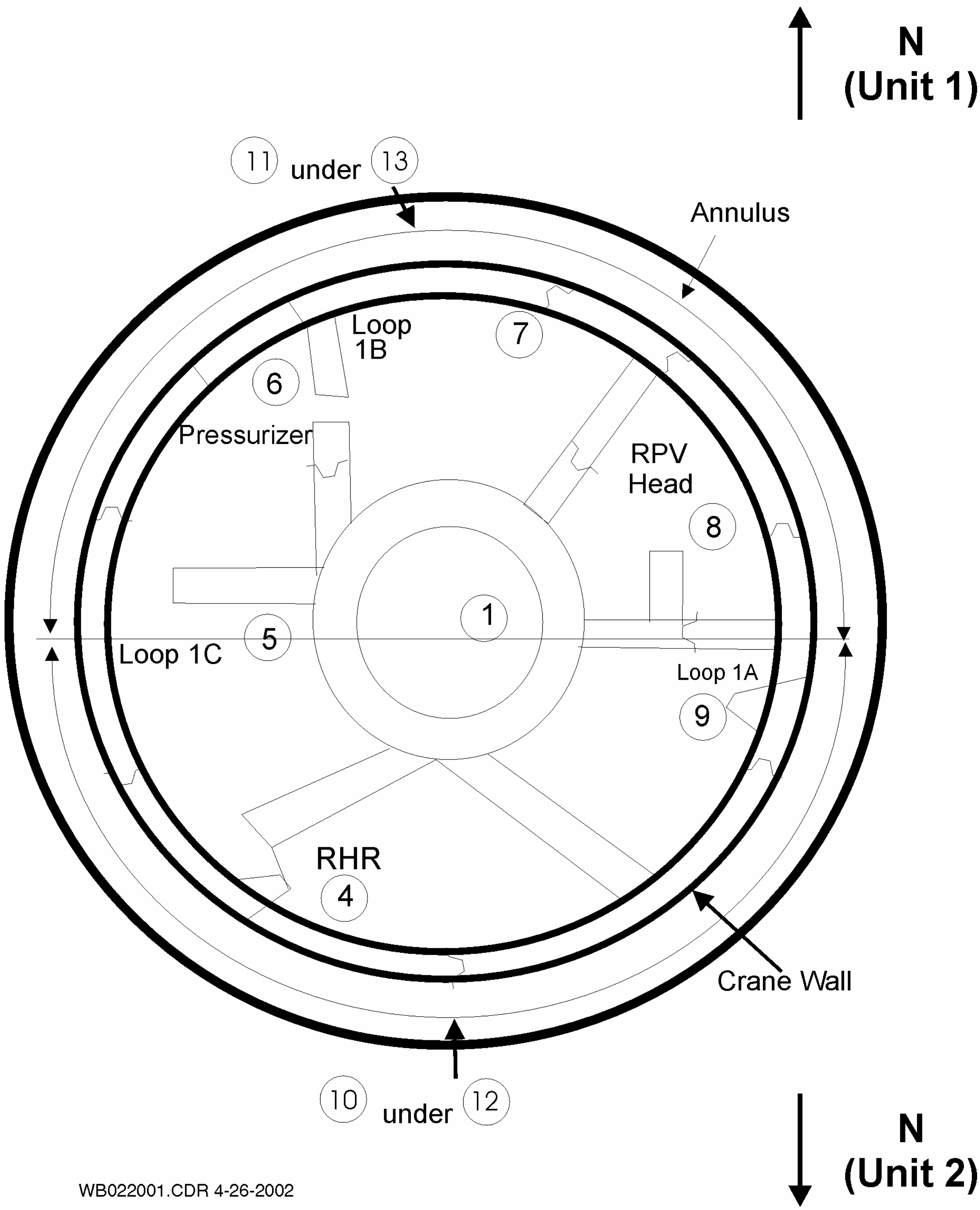
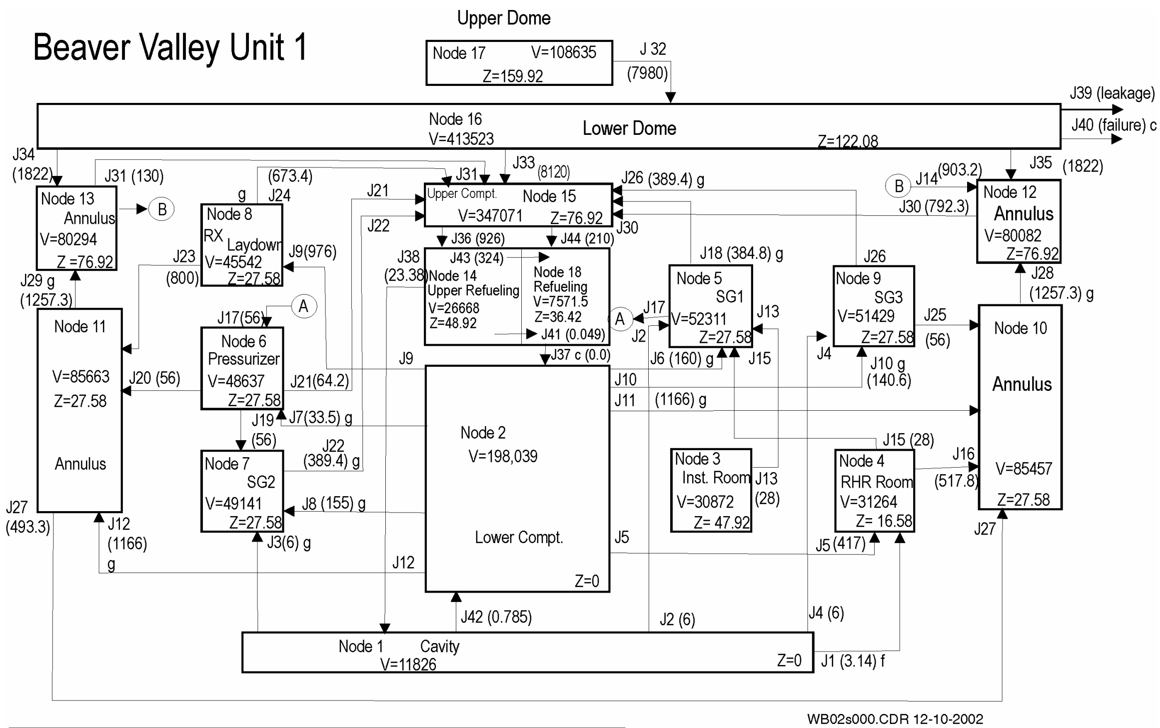


Figure 4-2
MAAP-DBA Containment Nodalization (Plan View) for BVPS-1 and -2



Notes:

(1) **Jn (A) g,c**
Jn: Junction number
(A): Junction flow area (ft.^{2})
g: Junction with grating and/or stairs
c: junction normally closed
f: junction opens on set ΔP

(2) Inside the node
V is volume (ft.^{3}) and
Z is elevation (ft.)

Figure 4-3a
MAAP-DBA Node and Junction Arrangement for BVPS-1

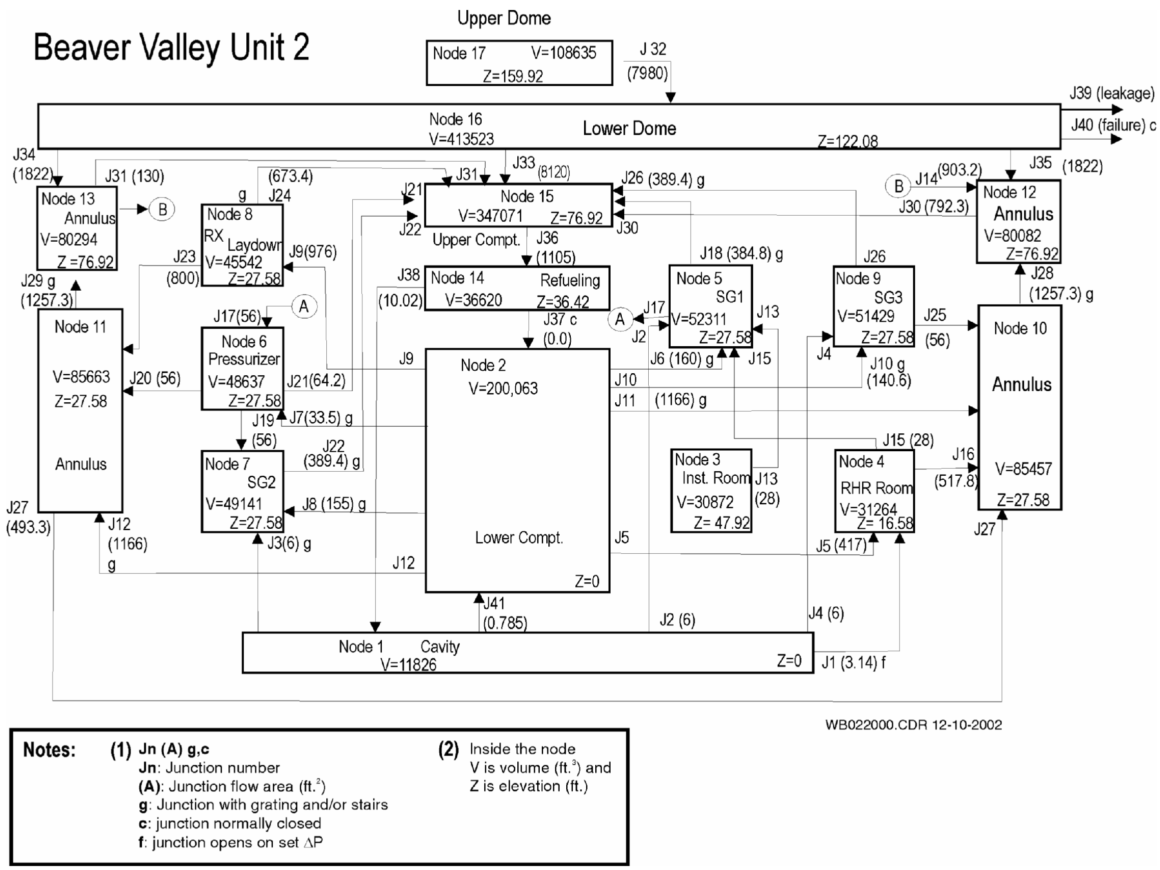
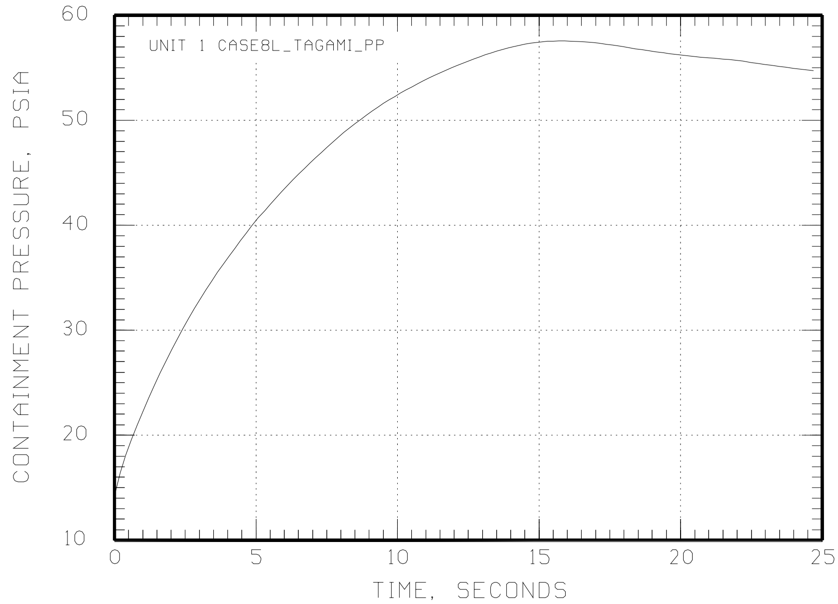


Figure 4-3b
MAAP-DBA Node and Junction Arrangement for BVPS-2



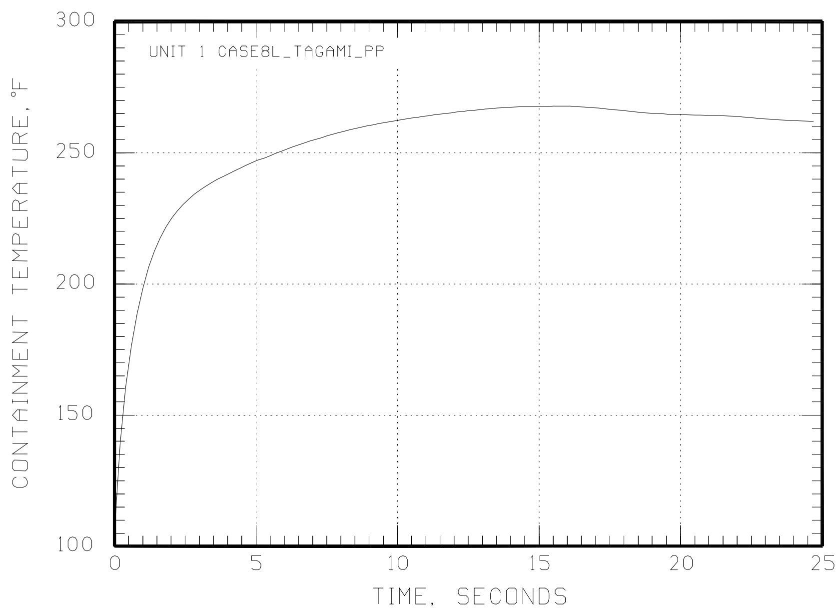


Figure 4-4
Composite Figure of the Containment Pressurization and Gas Atmosphere Temperature Histories for the DEHL Analyses for BVPS-1

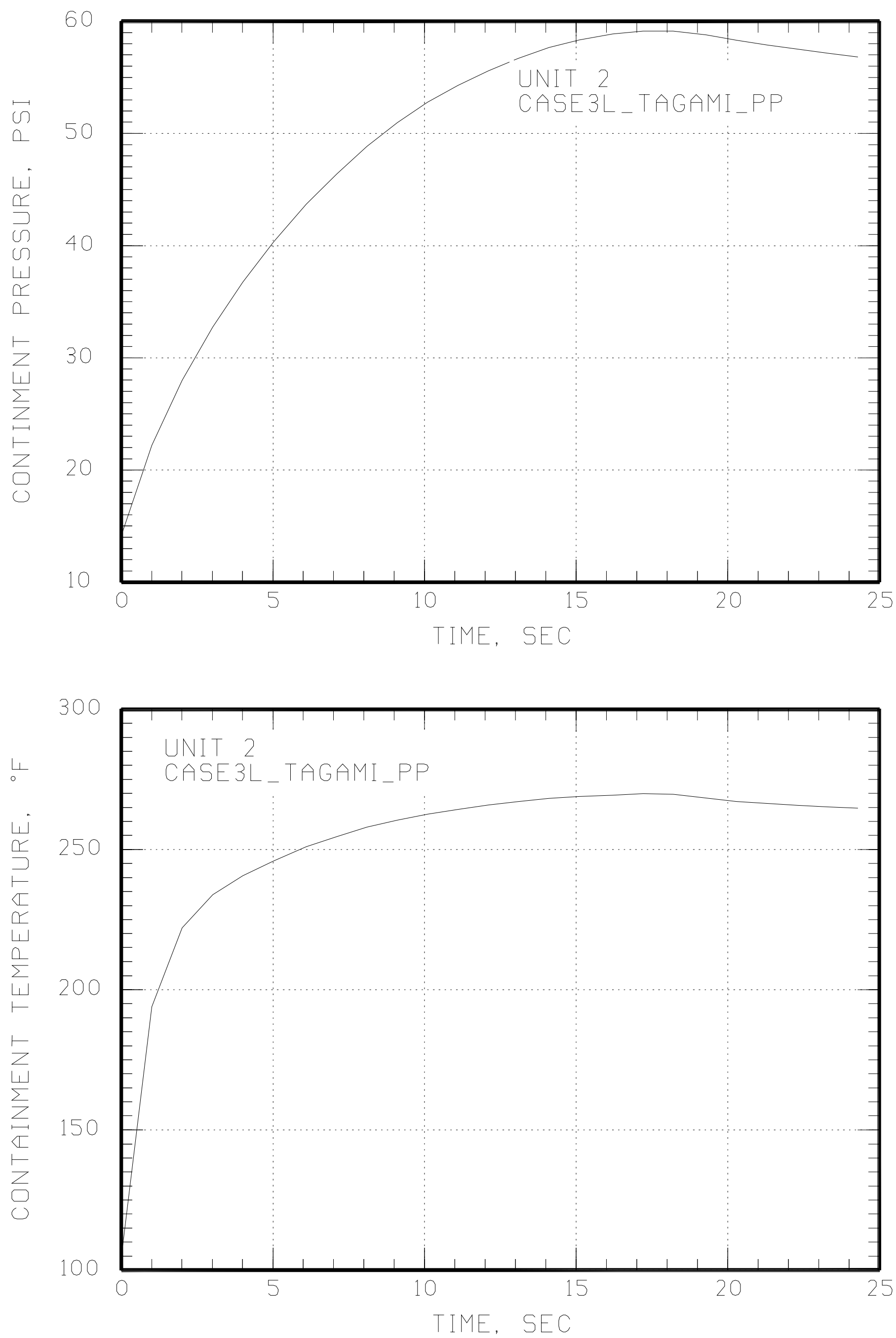


Figure 4-5
Composite Figure of the Containment Pressurization and Gas Atmosphere Temperature Histories for the DEHL Analyses for BVPS-2

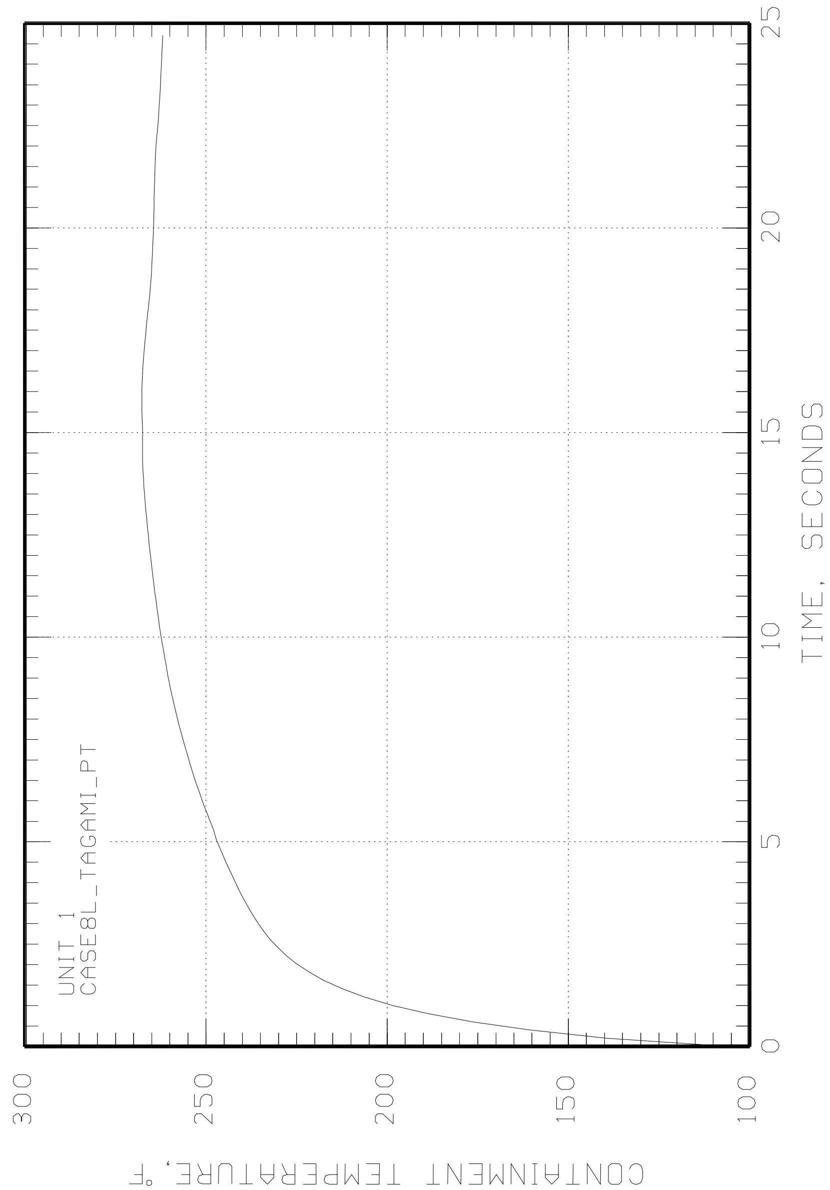


Figure 4-6
Gas Temperature for Case 8L (Double-Ended Hot Leg Break) for BVPS-1

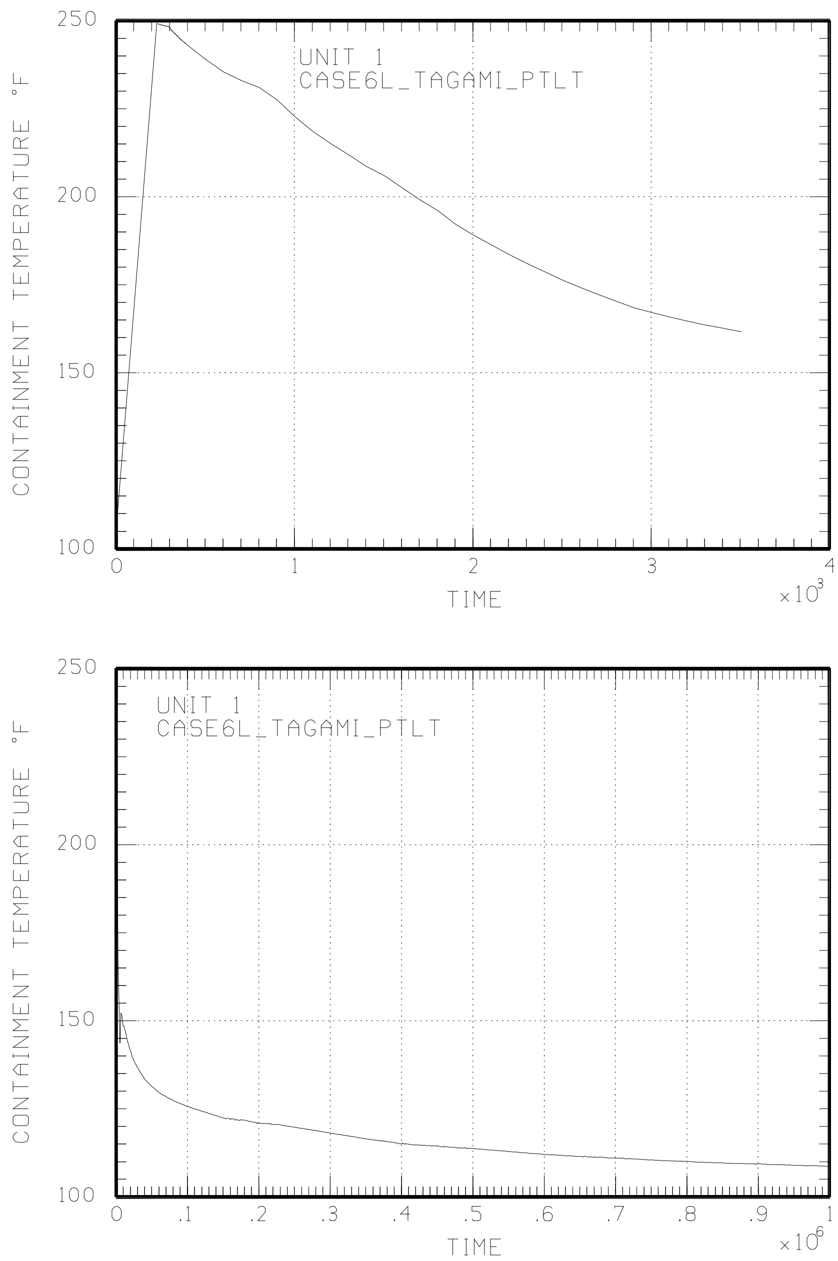


Figure 4-7
Gas Temperature for Case 6L (Double-Ended Pump Suction Break) for BVPS-1

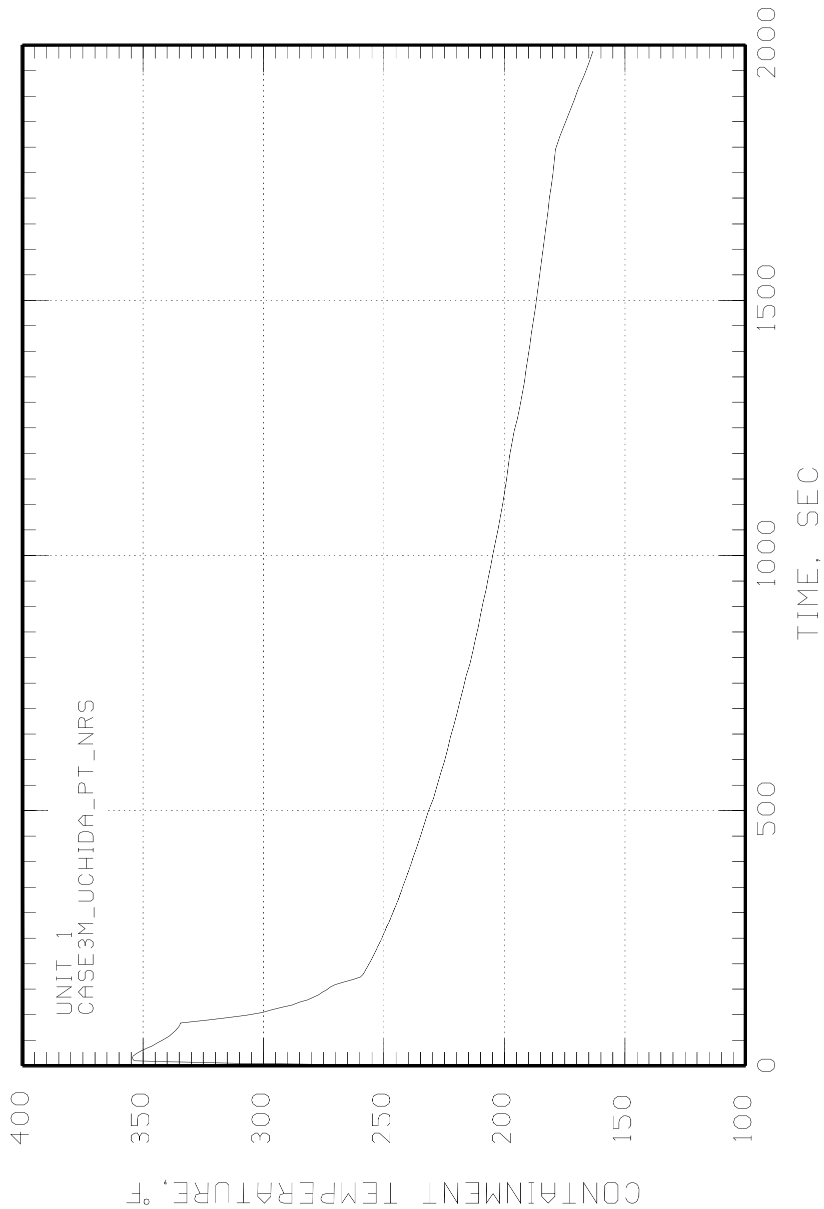


Figure 4-8
Gas Temperature for Case 3M for BVPS-1

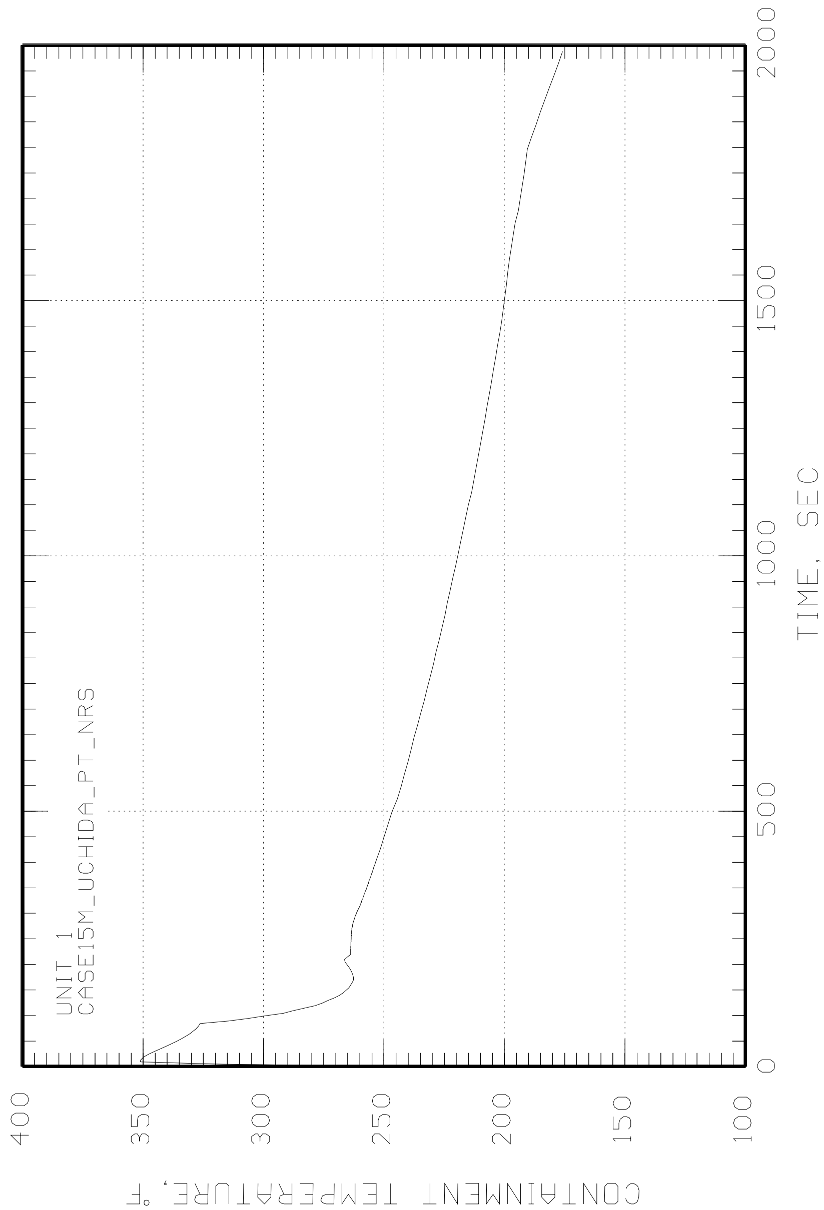


Figure 4-9
Gas Temperature for Case 15M for BVPS-1

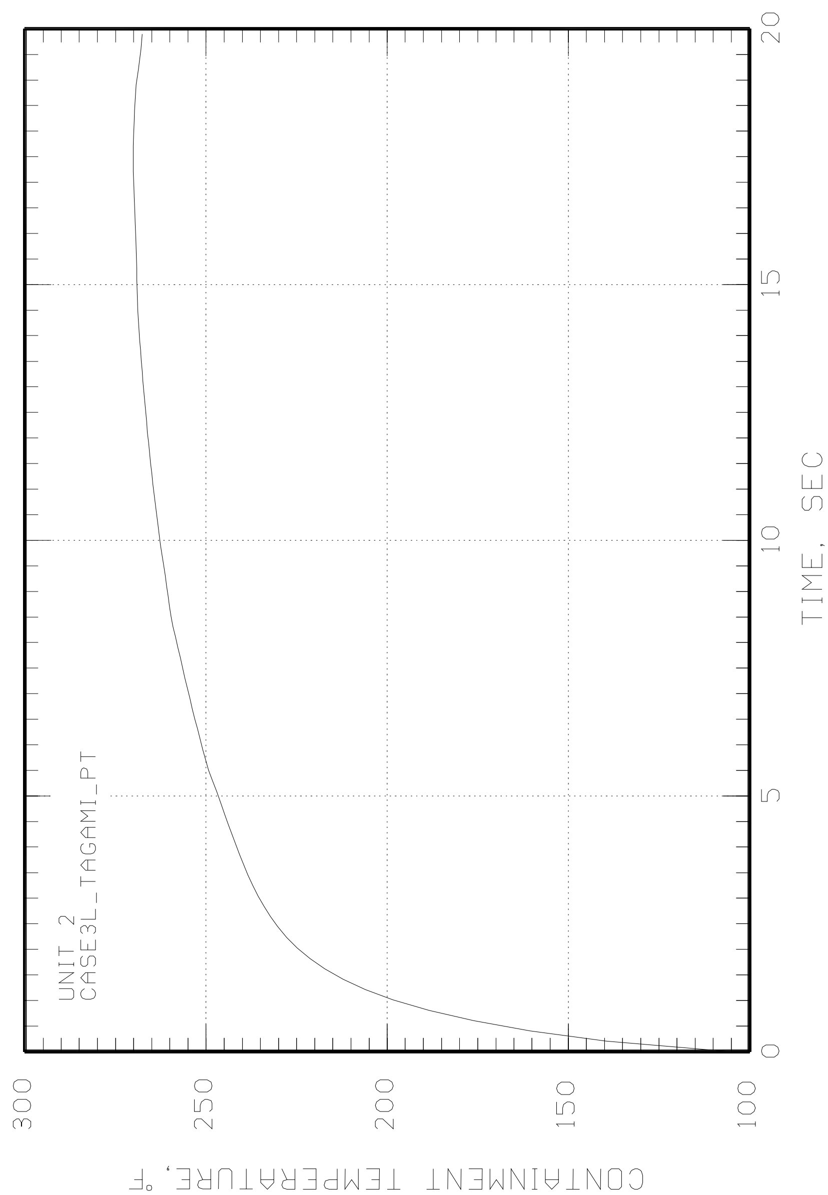


Figure 4-10
Gas Temperature for Case 3L (Double-Ended Hot Leg Break) for BVPS-2

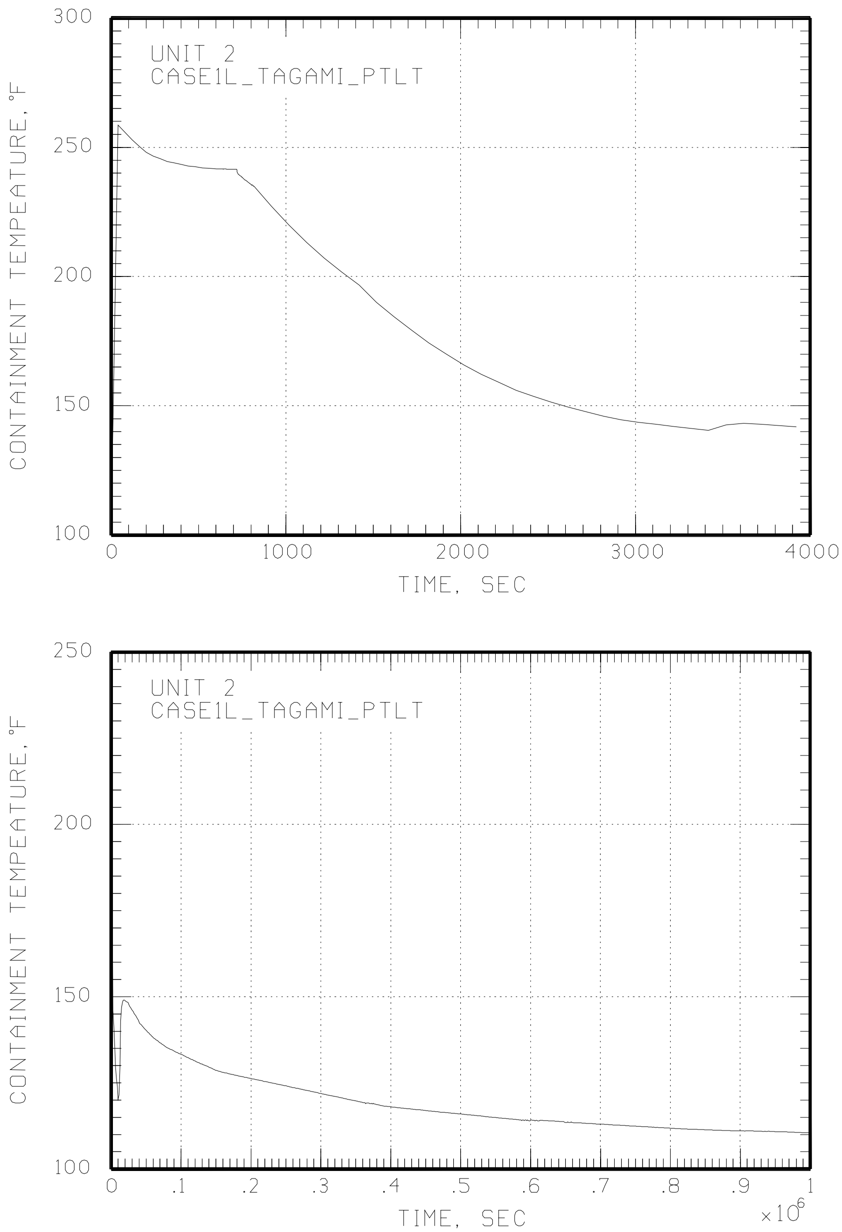


Figure 4-11
Gas Temperature for Case 1L (Double-Ended Pump Suction Break) for BVPS-2

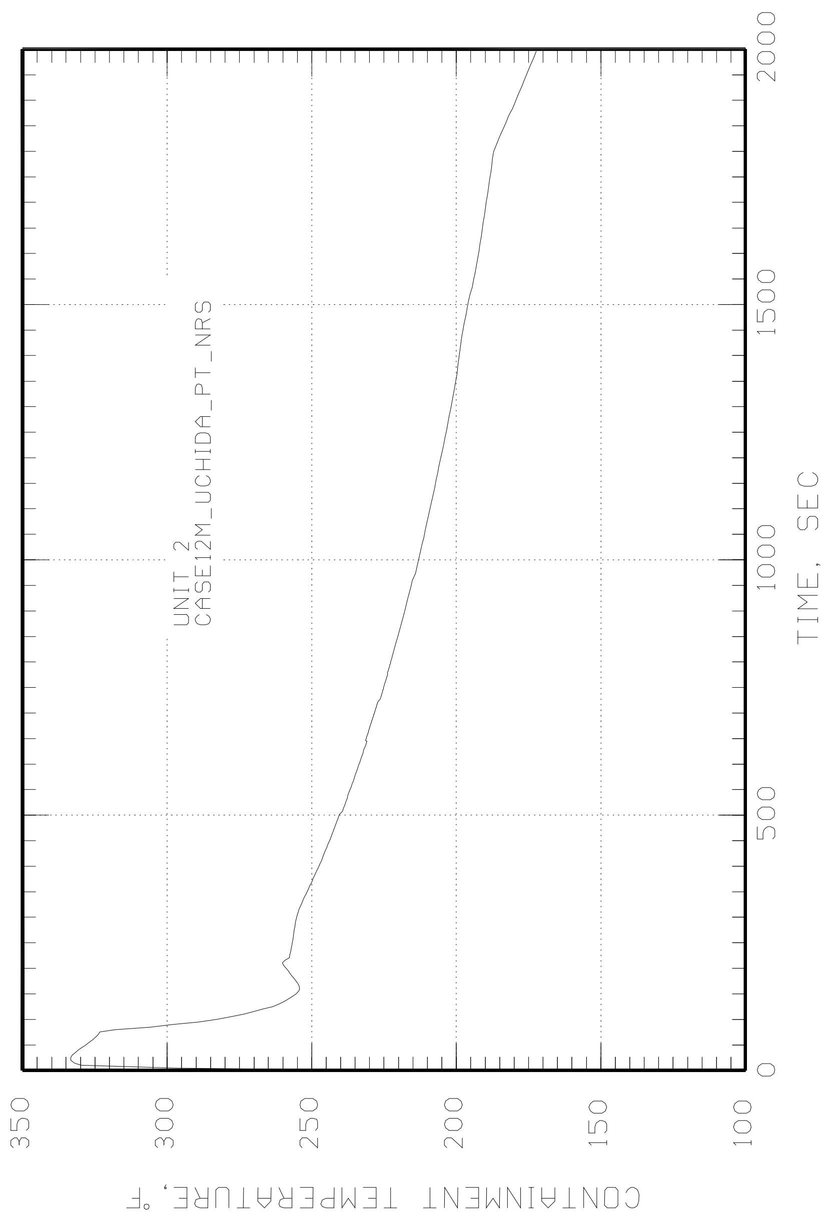


Figure 4-12
Gas Temperature for Case 12M for BVPS-2

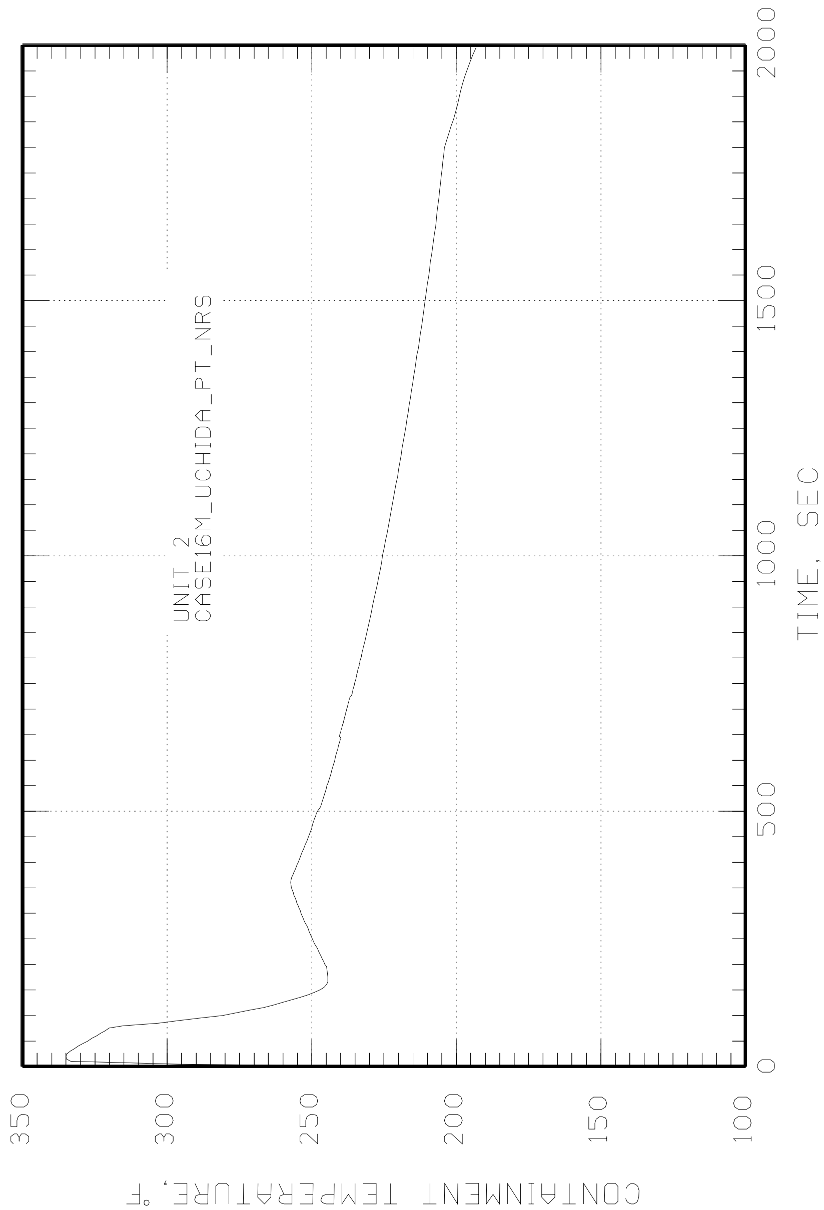


Figure 4-13
Gas Temperature for Case 16M for BVPS-2

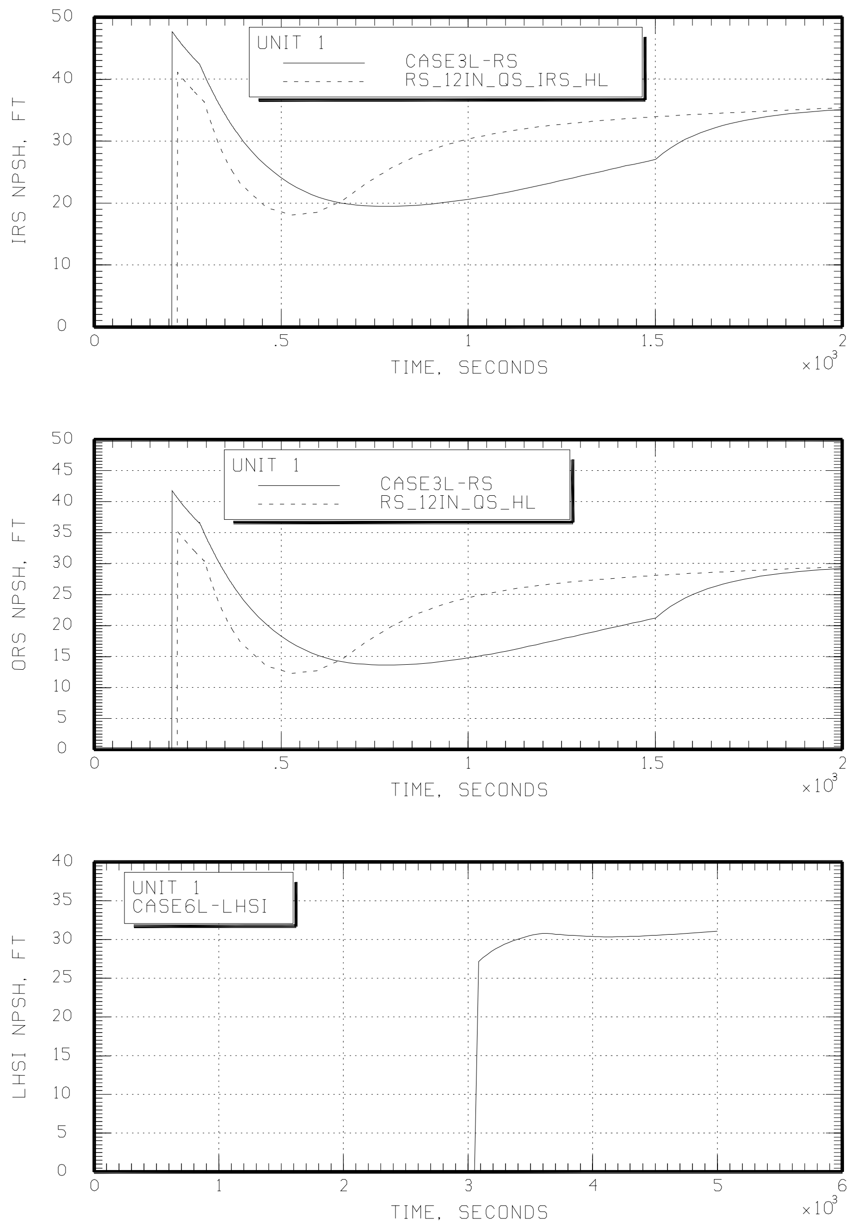


Figure 4-14
Available NPSH for BVPS-1 RS and LHSI Pumps

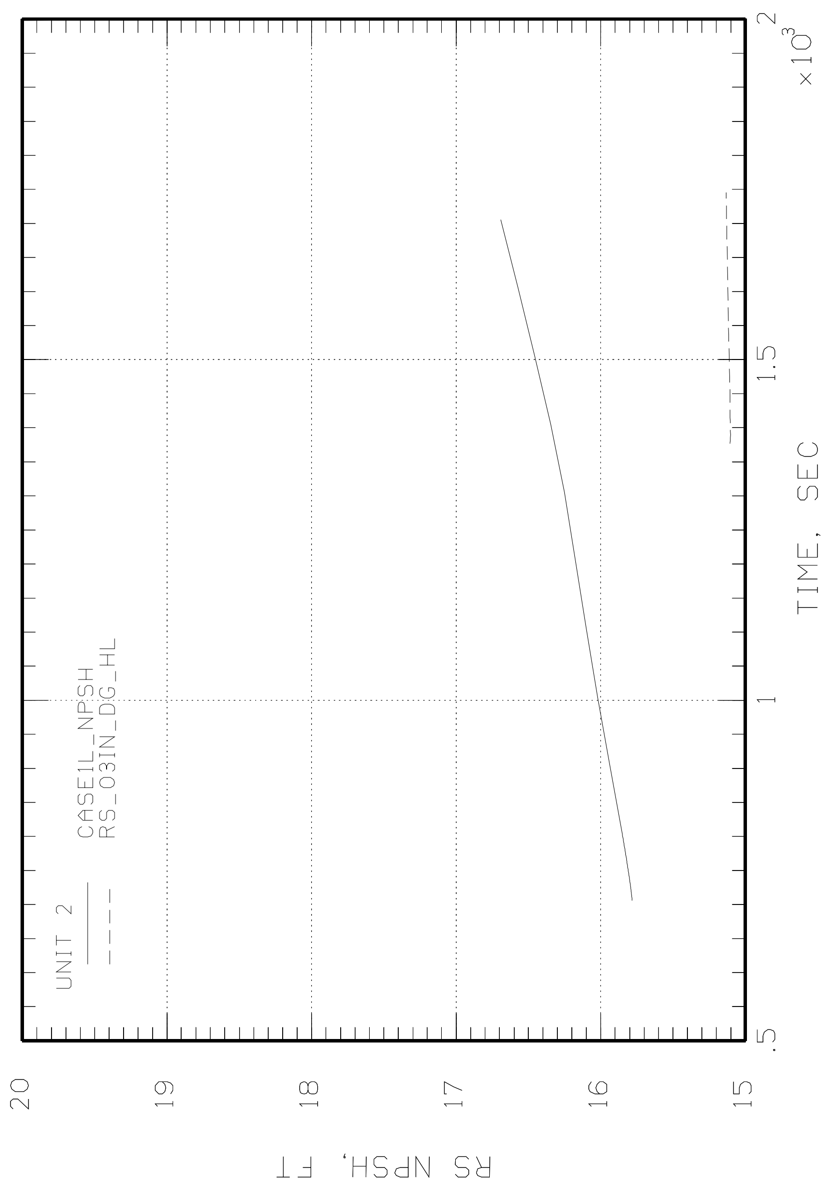


Figure 4-15
Available NPSH for BVPS-2 RS Pump

5 RADIOLOGICAL ASSESSMENT

5.1 INTRODUCTION

This chapter addresses the potential radiological impacts of changing the normal operating pressure in the BVPS-1 and -2 Containment Buildings from sub-atmospheric to atmospheric. Also included is an assessment of the radiological impact of eliminating the current post-accident design basis requirement of achieving sub-atmospheric conditions inside containment within one hour following accidents that result in containment pressurization. Note that to simplify documentation of this assessment; this chapter will refer to the above changes in design basis as atmospheric conversion.

The radiological assessments presented in this section are divided into two parts.

- Section 5.2 addresses the impact of atmospheric conversion on the post-accident radiological dose consequences at the site boundary (i.e., Exclusion Area Boundary and Low Population Zone), and at on-site locations such as the Control Room and the Emergency Response Facility (ERF).

- Section 5.3 addresses other radiological assessments; specifically, the impact of atmospheric conversion on normal plant operation, post accident radiological equipment qualification, operator exposure while performing mitigation functions in vital areas, and iodine loading on charcoal filters in safety related plant ventilation systems.

5.2 SITE BOUNDARY AND CONTROL ROOM/ERF DOSES

By Letter L-02-069, dated June 5, 2002 (Reference 14), which forwarded License Amendment Request (LAR) Nos. 300 (BVPS-1) and 172 (BVPS-2), and in accordance with 10CFR50.67 (Reference 1) and Standard Review Plan 15.0.1 (Reference 2), FENOC had proposed to revise the accident source term used in the BVPS-1 and -2 design basis site boundary and control room dose analyses, for those events impacted by atmospheric conversion, with the selective implementation of Alternative Source Terms (AST). As discussed in LAR Nos. 300 and 172, the design basis accidents that are impacted by atmospheric conversion are those that result in pressure transients in containment; i.e., the Loss of Coolant Accident (LOCA), and the Control Rod Ejection Accident (CREA). In accordance with plant design basis, the LOCA was analyzed for impact on the dose consequences at the site boundary, the control room and at the Emergency Response Facility (ERF). The CREA was analyzed for the impact on dose consequences at the site boundary and at the control room.

By letter L-03-038 (Reference 15), FENOC requested a phased approach to gain approval and implement LAR Nos. 300 and 172. The first phase was to gain approval and implement the control room habitability system Technical Specification changes for both BVPS units. Included in the first phase was the selective application of the Alternative Source Term (AST) methodology, and incorporation of the ARCON96 on-site atmospheric dispersion factors to determine the dose consequences of the LOCA and the CREA. Based on the technical justifications presented in FENOC Letter L-03-038, the proposed control room habitability system Technical Specification changes could be implemented regardless of whether the two BVPS units are operating at sub-atmospheric or atmospheric conditions.

NRC safety evaluation and approval of the selective implementation of Alternative Source Terms including use of atmospheric dispersion factors developed based on ARCON96, for the LOCA and the CREA, and the control room habitability technical specification changes at BVPS, was issued on September 10, 2003, via Amendment Nos. 257 and 139 to Facility Operating License Nos. DPR-66 and NPF-73, respectively. (Reference 16) As noted in Section 3.5.3 and 3.5.7 of the NRC SER, the containment leakage assumptions made in the LOCA and the CREA dose consequence analyses were intended to support a proposed containment conversion from sub-atmospheric to atmospheric operating conditions. In addition, and as noted in Section 3.1 of the NRC SER, the core inventory utilized for the dose assessments supported a licensed thermal power level of 2900 MWt to support applicability for the EPU power level as well as the current licensed power level.

Based on the above previously submitted assessments it is concluded that the radiological dose consequences at the site boundary, control room and ERF (LOCA only) remain within the regulatory limits of 10CFR50.67 following atmospheric conversion.

5.3 OTHER RADIOLOGICAL ASSESSMENTS

The assessments discussed below address the remaining potential radiological impacts of containment conversion.

5.3.1 Normal Operation Assessment

Containment operation at atmospheric conditions does not increase the source strength of any existing radiation source, nor does it result in additional radiation sources or release paths. Consequently, there is no impact on plant shielding requirements. Also, neither the effluent streams associated with liquid and solid radioactive waste are affected, nor are their treatment requirements.

The change to an atmospheric containment will result in a small decrease in the radioactive gaseous effluents from the containment during normal operations. While the gaseous radioactivity in containment will be eventually released to the environment via the containment vacuum system, this activity release will occur less frequently, resulting in a longer decay time for this effluent stream. Since the containment vacuum pump exhaust is not a major pathway, the net effect of atmospheric conversion will be a minor reduction in the offsite normal operation gaseous effluent dose.

In conclusion, an atmospheric containment during normal operations will not impact BVPS-1 or BVPS-2 compliance with the operator exposure limits set forth by 10CFR20, or with the public exposure limits set by 10CFR50 Appendix I (References 4 and 5).

5.3.2 Accident Assessments

To ensure a consistent design basis as it relates to radiological analyses impacted by atmospheric conversion, this section summarizes the impact of long-term post LOCA containment leakage resulting from atmospheric conversion and associated use of AST methodology on the:

- post-LOCA component of the total integrated dose used to evaluate equipment qualification of Class 1E equipment located in impacted areas.

- post-LOCA operator exposure while performing vital functions in areas impacted by the design change.

- iodine loading associated with the post-accident ventilation charcoal filters (i.e., the Supplementary Leak Collection and Release System (SLCRS) filter, the Control Room Post Accident Ventilation Air Intake filter, and the Emergency Response Facility Post Accident Ventilation Recirculation Filter).

5.3.2.1 Equipment Qualification

The equipment qualification (EQ) assessment takes into consideration the impact of long-term post-LOCA airborne radiation levels in the plant, and increased buildup of activity on the Supplementary Leak Collection and Release System (SLCRS) and control room HEPA and charcoal filters, due to atmospheric conversion/long-term containment leakage.

The potential impact on EQ doses due to these additional sources resulting from atmospheric conversion is assessed using AST assumptions and is conservatively based on EPU conditions. The increase in the post-LOCA component of the EQ dose resulting from atmospheric conversion, if any, is added to the existing dose estimates which are based on TID 14844 (Reference 7) source terms and reflect the current power level.

This approach is acceptable based on Section 1.3.5 of Regulatory Guide 1.183 (Reference 3) which indicates that though EQ analyses impacted by plant modifications should be updated to address the impacts, no plant modification is required to address the impact of the difference in source term characteristics (i.e., AST vs. TID 14844) on EQ doses.

Building Airborne Dose Contribution

Implementation of atmospheric conversion eliminates the current design basis of termination of containment leakage within one (1) hour post-DBA. An examination of areas contiguous to the containment was performed to analyze the impact of increased leakage time. Credit was taken for the operation of the SLCRS ventilation system that takes suction from the areas contiguous to the containment to minimize the spread of airborne radioactivity to adjacent areas non-contiguous to containment. For areas non-contiguous to containment, airborne gamma doses were calculated based on airborne concentrations that would be experienced if containment leakage was transported by atmospheric dispersion to that location without considering the exchange rate for building atmospheres with the environment. This maximized the potential integrated dose at each location analyzed.

Beta doses outside containment were not addressed since the radiation sensitive portions of safety related components are encased or otherwise protected from airborne radioactivity.

The impact of this review will be incorporated into the radiation zone maps during implementation of this design change. The study indicates that the impact of the airborne source due to atmospheric conversion on the post accident EQ doses in harsh environments is minimal, and that any increase in mild environments will not cause the zone to change from mild to harsh.

Impact of Increased Radioactivity Accumulation on SLCRS and Control Room Filters.

Implementation of atmospheric conversion no longer requires the termination of containment leakage within one (1) hour post-DBA. The post LOCA EQ dose estimates in the SLCRS and control room filter cubicles, and those cubicles adjacent to the filter cubicles, currently take credit for termination of containment leakage in 1 hour. Continued containment leakage over a period of 6 months will result in an increase in the post LOCA dose estimates in the referenced cubicles due to activity buildup on the filters.

BVPS-1 Assessment

Evaluation of the impact of increased activity accumulation on the SLCRS and control room filters at BVPS-1 indicates that radiation levels due to the accident contribution will result in an increase in the total integrated dose (normal plus accident) reported in the BVPS-1 EQ Radiation Zone Tables for the following areas:

- Elevation 768 ft-7 inches of the Auxiliary Building: Zones J (General Area) and I (SLCRS cubicle)
- Elevation 752 ft-6 inches of the Auxiliary Building: Zone H (General Area)
- Elevation 713 ft-6 inches of the Control Building: Air Conditioning Equipment Room (BVPS-1 Control Room Filter Cubicle)

BVPS-2 Assessment

Evaluation of the impact of increased activity accumulation on the SLCRS and control room filters at BVPS-2 indicates that radiation levels due to the accident contribution will result in an increase in the total integrated dose (normal plus accident) reported in the BVPS-2 EQ Radiation Zone Tables for the following areas:

- Elevation 773 ft-6 inches of the Auxiliary Building: Zone 80 (SLCRS cubicle)
- Elevation 735 ft-6 inches of the Control Building: Zone A (BVPS-2 Control Room Filter Cubicle; also called Fan Room)

Safety related equipment located in the referenced BVPS-1 and -2 EQ zones have been reviewed to ensure that they are qualified to the updated levels. Impact of atmospheric conversion on equipment qualification is addressed in Section 7.11.

5.3.2.2 Post LOCA Vital Access

The vital access assessment takes into consideration the impact of atmospheric conversion and associated use of AST source terms on the operator exposure while performing vital functions in areas impacted by the design change (specifically, areas adjacent to the SLCRS filters). The potential impact on vital access doses due to these additional sources resulting from atmospheric conversion is assessed using AST assumptions and is conservatively based on uprated conditions. The increase in operator exposure resulting from atmospheric conversion, if any, is added to the existing dose estimates, which are based on TID 14844 source terms and reflect the current power level.

This approach is acceptable based on the AST bench marking study reported in SECY-98-154 (Reference 17) which concluded that results of analyses based on TID 14844 would be more limiting earlier on in the event, after which time the AST results would be more limiting. The NRC SER for Fort Calhoun Station's implementation of the AST (Reference 19), referenced the SECY-98-154 study as the source for the conclusion that results of analyses based on TID 14844 would be more limiting for periods up to one to four months, after which time the AST results would be more limiting. Post-LOCA access to vital areas, usually occur, within the first one or two weeks when the original TID 14844 source term is more limiting.

The original NUREG-0737 II.B.2 (Reference 8) reviews of post-LOCA vital area access were intended to determine the adequacy of plant shielding and therefore focused on "contained" sources. Therefore, airborne levels in the plant due to long term containment leakage resulting from atmospheric conversion are not addressed.

The BVPS licensing basis for vital access is based NUREG-0737 II.B.2 and considers "contained" sources as documented for BVPS-2, in BVPS-2 UFSAR Section 12.3.2.10 and the NRC acceptance of this approach in the BVPS-2 SER (Reference 9); and for BVPS-1, in the NRC SER (Reference 10) which confirms acceptability of the methodology utilized in the vital access assessment documented in Reference 11.

Impact of Increased Radioactivity Accumulation on SLCRS Filters.

Evaluation of the impact of increased activity accumulation on the SLCRS filters at BVPS-1 and -2 on the maximum dose rate in areas adjacent to the filter cubicle on the same floor, as well as the floor below, for the time period between T=0 to T=30 days, is developed using AST methodology and the LOCA model used in the dose consequence analyses, with the following exception. All of the containment leakage is processed through the SLCRS filters with the assumption that the filters are 100% efficient. The maximum dose rate at a representative location outside the cubicle due to the SLCRS filters occurs at approximately T=3 hrs, and is estimated to be bounded by 0.2 Rem/hr at either unit.

<u>BVPS-1 Assessment</u>

As documented in the NRC SER issued to BVPS-1 relative to compliance with NUREG-0737 II.B.2, (Reference 10), the BPVS-1 licensing basis does not include estimated doses per operator mission; rather it is a documented evaluation of the worst case post accident dose rates in plant areas that may need access following a LOCA, and identification of shielding inadequacies that were satisfactorily resolved.

In accordance with the referenced SER, NRC acceptance of the BVPS-1 submittal included the availability of procedural controls implemented for post accident vital access which include figures that depict primary and alternate access routes and associated area dose rates to perform various post accident operations.

Review of BVPS-1 Health Physics Procedure REOP 2.1 (Reference6) indicates that Route No. EWT 4 (access to the SA-9/10 Monitors) and Route No. EWT 5 (access to the Sping-4 Monitors) are the only access paths that pass through areas where the radiation levels maybe impacted by the increased radioactivity buildup on the SLCRS filters due to atmospheric conversion and resulting long-term containment leakage. As shown in Figures 5.2.1.7 through 5.2.1.10 of REOP 2.1, both routes involves passage through the general area outside the SLCRS filter cubicle on El 768'7"of the Auxiliary Building and on the floor below, i.e., El 752 ft-6 inches.

A review of the estimated dose rates identified in Figures 5.2.1.7 through 5.2.1.10 of REOP 2.1 indicate that the existing worst case values (from all sources) in the general area outside the SLCRS cubicle ranges from 11 Rem / hr to 38 Rem / hr, and the existing calculated dose rate in the area below the SLCRS cubicle ranges from 5 Rem / hr to 22 Rem / hr.

The incremental contribution from the SLCRS filters following atmospheric conversion, 0.2 R/hr (estimated at T=3 hrs), is insignificant compared to the existing calculated dose rates. Atmospheric conversion will therefore have a negligible impact on the vital access dose rate estimates presented in Figures 5.2.1.7 through 5.2.1.10 of REOP 2.1.

It is therefore concluded that BVPS-1 remains in compliance with NUREG 0737 II.B.2 relative to operator mission doses post-LOCA, considering an atmospheric containment.

<u>BVPS-2 Assessment</u>

Review of the BVPS-2 vital access paths outlined in UFSAR Section 12.3.2.10 indicates that Access Routes 13 (obtaining and analyzing a post-accident effluent sample at T=1 hr) and Route 14 (re-energizing ECCS valve circuits at T=25 minutes) are the only access paths that pass through areas where the radiation levels may be impacted by the increased radioactivity buildup on the SLCRS filters due to atmospheric conversion and resulting long term containment leakage. As shown in Figures 12.3-34 through 12.3-36 of the BVPS-2 UFSAR, Route 13 involves passage through the general area outside the SLCRS filter cubicle on El 773 ft-6 inches of the Auxiliary Building, whereas Route 14 involves passage on the floor below, i.e., El 755 ft-6 inches.

Since the required time for access for both routes is within the first hour, atmospheric conversion will have no impact on the operator dose estimates reported in UFSAR Table 12.3-4 for the referenced routes.

The following evaluation is provided in the event that access occurs after T=1 hr.

Review of the analyses supporting the BVPS-2 operator dose estimates presented in UFSAR Table 12.3-4 for the referenced routes indicated that the existing T=1 hr dose rate (from all sources) in the general area outside the SLCRS cubicle ranges from 7 Rem/hr to 23 Rem/hr, and the existing calculated dose rate in the area below the SLCRS cubicle is 3.4 Rem/hr at T=25 min.

The worst case incremental dose rate from the SLCRS filter has been calculated to be 0.2 Rem/hr (which occurs at approximately T=3 hrs). This increase is insignificant compared to the existing calculated doses, and supports the conclusion that atmospheric conversion will have a negligible impact on the vital access dose estimates presented in the BVPS-2 UFSAR.

It is therefore concluded that BVPS-2 remains in compliance with NUREG 0737 II.B.2 relative to operator mission doses post-LOCA, considering an atmospheric containment.

5.3.2.3 Iodine Loading on Accident Filters

An evaluation was performed to determine the impact of the post-LOCA iodine loading on the Control Room, SLCRS and ERF charcoal filters following containment atmospheric conversion. With atmospheric conversion of the containment, leakage from the containment airspace is not terminated after an hour. The efficiency for the charcoal filters is dependent on maintaining the iodine loading within the design requirements established by Regulatory Guide 1.52 (Reference 13).

During a LOCA, iodine is released to the environment via containment atmospheric leakage or from sump liquid leakage via the ECCS systems or the RWST. The evaluation to determine the iodine buildup on the filters utilized the LOCA model used in the dose consequence analyses with several additional conservative assumptions as summarized below.

Iodine released into either the containment atmosphere or sump liquid was assumed to occur instantaneously and independently, using AST release fractions. This maximized the availability of iodine for release to the environment via either pathway. No credit was taken for depletion of iodine in the containment atmosphere via removal mechanisms such as sprays. All iodine was assumed to be either elemental or organic. No iodine decay was credited. Dispersion coefficients (χ/Q) from the containment and other release points were assumed to be the 0-2 hour values. To maximize filter inventory, 100% filter efficiency was assumed. If a charcoal filter existed on a source pathway prior to the filter being analyzed (for example, the SLCRS filter for ECCS or containment atmospheric leakage), it was not credited for iodine removal and a 0% filter efficiency was assumed for that filter. For RWST releases, the start of leakage release fraction (5x10-3per day) was used over the entire 30-day event. The mass of radioactive Tellurium released from the core was instantaneously added to the mass of iodine available for release in the containment atmosphere or sump liquid. This is conservative, since the decay of Tellurium is not instantaneous and branching fractions exist which would further reduce the mass of Tellurium converted to iodine.

The BVPS core iodine mass in grams is based on a core power level of 2918 MWt and presented below.

Nuclide	Core Iodine Mass (gram)
I127	3.63E+03
I128	1.72E-02
I129	1.62E+04
I130	1.06E-00
I130M	6.86E-03
I-131	6.27E+02
I-132	1.09E+01
I-133	1.41E+02

Nuclide	Core Iodine Mass (gram)
I-133M	1.30E-03
I-134	6.64E-00
I-134M	4.17E-02
I-135	4.31E+01
I136	7.02E-02
I136M	1.92E-02
I137	2.22E-02
I138	3.01E-03
I139	5.17E-04
Total	**2.07E+04**

In accordance with Section 3I of RG 1.52 "The absorption unit should be designed for a maximum loading of 2.5 mg of total iodine (radioactive plus stable) per gram of activated carbon." The BVPS assessment determined that the iodine loading on the ESF ventilation system filters was as follows:

BVPS-1 Charcoal Filter Iodine Loading			
	SLCRS	**CR**	**ERF**
Charcoal Loading (mg I/gm C)	0.075	$1.3X10^{-3}$	$1.2X10^{-5}$

BVPS-2 Charcoal Filter Iodine Loading			
	SLCRS	**CR**	**ERF**
Charcoal Loading (mg I/gm C)	0.033	$5.77X10^{-4}$	$1.78X10\text{-}^{5}$

These values are well below the RG 1.52 acceptance criterion of 2.5 $\frac{\text{mg I}}{\text{gm Charcoal}}$.

5.4 CONCLUSIONS

The radiological analyses and evaluations documented in this section demonstrate that conversion of BVPS-1 and -2 containments to atmospheric will not impact compliance with applicable regulatory radiological dose limits for normal operation and for accidents, at the site boundary or in vital access areas, including the control rooms and the ERF. The evaluations also demonstrate continued compliance with the unit specific licensing commitments associated with NUREG 0737 II.B.2, the limitations on charcoal filter loading in accordance with Regulatory Guide 1.52, and the post-accident radiological environmental qualification requirements for class 1E electrical equipment in accordance with 10CFR50.49. (Reference 18)

5.5 REFERENCES

1. 10CFR50.67, "Accident Source Term."

2. NUREG-0800, Standard Review Plan 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms," Revision 0.

3. Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

4. Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

5. Code of Federal Regulations, Title 10, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion As Low As Reasonably Achievable for Radioactive Material in Light Water Cooled Nuclear Power Reactor Effluents."

6. BVPS-1 Health Physics Manual REOP 2.1, Revision 2, "Access and Dose Control for Vital Area Operations during Emergency Situations."

7. TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," 1962.

8. NUREG-0737, "Clarification of TMI Action Plan Requirements," Nov. 1980.

9. NUREG-1057, "Safety Evaluation Report related to the Operation of Beaver Valley Power Station BVPS-2," October 1985.

10. Safety Evaluation Report, "NUREG-0737, Item II.B.2 – Design Review of Plant Shielding – Corrective Actions for Access to Vital Areas, Beaver Valley Power Station Unit No. 1," date November 8, 1982.

11. Nuclear Services Corporation (A Division of QUADREX Corporation) Report No. QUAD-1-8-040, "Design Review of Plant Shielding of Spaces for Post-Accident Operation," Revision 1, April 25, 1980.

12. NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements – Requirements for Emergency Response Capability," December 17, 1982.

13. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2.

14. FENOC Letter L-02-069 to NRC dated June 5, 2002, "Beaver Valley Power Station Units No. 1 and 2, Licensing Amendment Request Nos. 300 and 172."

15. FENOC Letter L-03-038 to NRC dated March 31, 2003, “Beaver Valley Power Station Units No. 1 and 2, Licensing Amendment Request Nos. 300 and 172 Implementation Plan.”

16. NRC Safety Evaluation Report enclosing Amendment No. 257 (BVPS-1) and No. 139 (BVPS-2), “Beaver Valley Power Station Units 1&2 – Issuance of Amendment Re: Selective Implementation of Alternate Source Terms and Control Room Habitability Technical Specification Changes,” September 10, 2003.

17. SECY-98-154, “Results of the Revised (NUREG-1465) Source Term Rebase lining for Operating Reactors,” June 30, 1998.

18. Code of Federal Regulations, Title 10, Part 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”

19. NRC SER for Fort Calhoun Station, Amendment No. 201 to Facility Operating License, dated December 5, 2001.

6 OTHER CONSIDERATIONS

The preceding chapters of this report describe the analyses performed to evaluate and justify the acceptability of converting the containment pressure design basis from sub-atmospheric to atmospheric, including:

- Chapter 2 describes the analysis performed to define PCWG parameters for use as input to the LOCA and MSLB mass and energy release analysis.

- Chapter 3 describes the analysis performed to develop LOCA and MSLB mass and energy release data for use as input to the containment analysis.

- Chapter 4 describes the containment analysis performed to justify conversion of the containment pressure design basis from sub-atmospheric to atmospheric.

- Chapter 5 describes the radiological dose analysis performed to assess and justify the acceptability of radiological doses following conversion of the containment pressure design basis to atmospheric.

The inputs and assumptions to the above analyses were selected to support the future EPU of BVPS-1 and -2. In addition, evaluations were performed for the above areas to show that the analyses performed for the EPU are applicable to or bound the current power level. These evaluations were performed to justify the potential implementation of containment conversion at the current power level (i.e., prior to implementation of EPU) even though the containment conversion analyses described above were performed based on inputs and assumptions for the EPU.

This chapter describes the evaluations performed for other considerations that are potentially impacted by containment conversion to assess and justify that various design and licensing requirements will continue to be satisfied should containment conversion and associated plant equipment changes be implemented at the current power level. Similarly, Chapter 7 describes the analyses and evaluations performed for Programs developed or implemented by plant personnel to demonstrate that topical areas will continue to comply with various design and licensing requirements following containment conversion at the current power level. Chapter 8 describes the environmental assessment performed to support containment conversion at the current power level. Finally, Chapter 9 describes the MAAP-DBA methodology and experiments. The EPU includes analyses and evaluations for these areas to support implementation of EPU after or concurrent with implementation of containment conversion. These EPU project analyses will be described in a separate report and submittal.

6.1 UFSAR ACCIDENT ANALYSES

The majority of the changes associated with containment conversion as summarized in Section 1.1 do not adversely impact the UFSAR accident analyses performed at the current power level; however, the following changes do have a potential impact and were evaluated in support of containment conversion:

- Increase in the containment pressure range for normal operation (i.e., 12.8 to 14.2 psia).

- Increase in the containment pressure related safety analysis limits (SALs) and the Engineered Safety Feature Actuation System (ESFAS) setpoints for high, intermediate high-high, and high-high containment pressure trips.

- Decrease in the ESFAS setpoint for RWST low level switchover from the injection to the recirculation mode (i.e., switchover of the recirculation spray pumps (BVPS-1 and -2) and low head safety injection pumps (BVPS-1) suction from the RWST to the containment recirculation sump).

- Decrease in the containment minimum temperature for normal operation.

- Elimination of the Quench Spray flow cutback logic on RWST low level (BVPS-1 only).

- Installation of flow limiting cavitating venturis in the AFW flow delivery lines to each steam generator (BVPS-1 only, venturis already exist on BVPS-2).

- Installation of fast acting flow isolation valves in the main feedwater lines to each steam generator (BVPS-1 only, flow isolation valves already exist on BVPS-2).

6.1.1 LOCA Transients

The current containment designs for BVPS-1 and -2 incorporate a sub-atmospheric design, and the existing LOCA analyses assume that the initial containment pressure is within the sub-atmospheric containment technical specification limits as described in Technical Specification 3/4.6.1.4. The LOCA accident analyses are described in Section 14.3 of the BVPS-1 UFSAR, and in Section 15.6 of the BVPS-2 UFSAR.

An evaluation was performed for the existing LOCA analyses, assuming conversion of the design from a sub-atmospheric pressure to atmospheric pressure. Among the changes due to the conversion, the following list of changes was determined to be applicable to LOCA and have been evaluated to assess the impact on LOCA analyses:

- normal operation of the containment between 12.8 and 14.2 psia
- Containment Pressure-High SAL range of 18-22 psia
- decrease in the ESFAS RWST Level-Low switchover setpoint
- minimum containment temperature of 70°F
- elimination of Quench Spray flow cutback logic on RWST Level – Low (BVPS-1 only)
- installation of fast acting feedwater isolation valves (BVPS-1 only)
- installation of a cavitating venturi in each auxiliary feedwater injection line (BVPS-1 only).

The evaluation concludes that the LOCA analyses results are not adversely affected by the changes being proposed in connection with the conversion. Therefore, the Containment Conversion is acceptable with respect to the LOCA analyses of record.

6.1.1.1 Large Break LOCA

The BVPS-1 and -2 Large Break LOCA (LBLOCA) Analysis of Record (AORs) model minimum containment pressures of 8.9 psia and 9.0 psia, respectively. Westinghouse Appendix K LBLOCA analyses have shown increases in Peak Clad Temperature (PCT) for decreases in containment pressure, which is consistent with the concept of steam binding during reflood. Steam binding involves the volumetric displacement of steam with liquid. ECCS liquid enters the core and leaves as steam with entrained fluid. This change of thermodynamic state causes a significant decrease in the density between the entering fluid and the exiting steam. The large volumes of generated steam must be vented through the break in order to allow further reflooding of the core. When the containment pressure is decreased, the core pressure decreases, and ultimately this causes a further decrease in vapor density. The lower steam density means even more volume must be displaced per unit mass of liquid entering the core. The impact of steam binding leads to the requirements of conservatively low containment pressure during reflood. Therefore, the current LBLOCA AORs performed with minimum (sub-atmospheric) containment pressure bound operation at atmospheric pressure or higher for BVPS-1 and -2.

The increase in containment pressure naturally results in an increase in the Containment Pressure – High (High-1) setpoint to a maximum value of 22 psia. After accident initiation time, the RCS rapidly depressurizes past the Pressurizer Pressure – Low SI setpoint soon after pressurizing the containment to the Containment Pressure – High SI setpoint. Since the increase in containment pressure and High-1 SAL are very small compared to the initial RCS pressure, the change will have a negligible effect on the time at which the Pressurizer Pressure – Low or High-1 setpoint is reached. Since the time at which the SI signal is generated does not change significantly, the AOR is not affected.

The LBLOCA methodology uses nominal containment temperature. Reducing the minimum allowable temperature to 70°F does not impact the LBLOCA AOR. To conservatively reduce the containment pressure, the LBLOCA methodology also assumes maximum containment spray flow. So, eliminating the Quench Spray flow cutback logic on RWST Level-Low does not impact the results of the AOR. Additionally, the LBLOCA methodology does not model RWST volume or level, auxiliary feedwater or main feedwater. Therefore, the reduction in the RWST Level-Low setpoint at both units and the installation of the cavitating venturis in the AFW lines and fast acting flow isolation valves in the MFW lines at BVPS-1 have no impact on the LBLOCA AOR.

6.1.1.2 Small Break LOCA

An evaluation of the Small Break LOCA (SBLOCA) was performed to support the Containment Conversion.

In a SBLOCA the break in the cold leg spills to the containment. The small break sizes, coupled with the large pressure differential between the RCS and the containment, result in a choked flow (also called sonic flow) condition where the sub-cooled, two-phase or vapor flow through the break is at sonic velocity. The Modified Zaloudek, Moody break flow and Murdock-Baumann correlations used in the AOR NOTRUMP transients assume choked flow conditions during the transient time of interest and therefore are independent of the downstream (containment) pressure or temperature. Therefore, changing containment pressure from sub-atmospheric to atmospheric pressure has no impact on the BVPS-1 and -2 SBLOCA analyses.

Decreasing the RWST Level-Low switchover setpoint has the net effect of increasing the deliverable volume in the SBLOCA analysis. Increasing the deliverable volume results in a later switchover to sump recirculation. After switchover, the pumped ECCS water is assumed to be at a higher enthalpy than the RWST water. Increasing the time at which this is assumed to occur is beneficial to the SBLOCA transient. Therefore, the effects of decreasing the RWST Level-Low switchover setpoint have been bounded by the analysis of record. Part of the SBLOCA RWST draindown calculation assumes maximum containment spray flow rates to conservatively drain the RWST as rapidly as possible. As such, elimination of the Quench Spray flow cutback on RWST Level-Low switchover signal has no impact on the SBLOCA analyses.

The overall effect of installing cavitating venturis on plant parameters important to SBLOCA analysis is a corresponding drop in assumed AFW flow. An evaluation concluded that this drop in AFW flow would not result in any uncovery of the steam generator tubes on the secondary side. As such, the reduction in flow will not significantly alter the steam generator primary-to-secondary heat transfer properties during the SBLOCA transient. Therefore, the reduction in AFW flow due to the installation of cavitating venturis at BVPS-1 will have negligible impact on PCT.

Currently, the Westinghouse Small Break LOCA methodology assumes that delayed feedwater isolation is conservative and is accomplished for BVPS-1 and -2 by the Feedwater Regulating Valves. Installation of the fast acting valves is a redundant system and has no impact on the SBLOCA AOR.

The Small Break LOCA methodology does not model containment temperature or the Containment Pressure – High setpoint. Therefore, these changes do not impact the SBLOCA AOR.

6.1.1.3 Hot Leg Switchover

The LOCA related Hot Leg Switchover (HLSO) analysis employs primarily mass based calculations, while input data is often only available in volumetric units. The local temperature and pressure are used with the ASME steam tables to provide temperature/pressure dependent physical property values used to convert from volumetric to mass based units (density, enthalpy…). Because liquid is incompressible, and all of the relevant initial condition calculations assume sub-cooled or saturated water at containment pressure, the small change in pressure assumed for the containment has a negligible effect on those physical parameters used in the units conversion. In addition, the hot leg switchover time does not change as the result of containment atmospheric conversion. Therefore, changing from sub-atmospheric to atmospheric containment pressure or reducing containment temperature has a negligible effect on the Hot Leg Switchover for BVPS-1 and -2. The BVPS Hot Leg Switchover Time calculation assumes maximum deliverable RWST volume. This typically includes the entire RWST volume, including RWST piping. Therefore, the increase in RWST deliverable volume due to the decrease in the RWST Level – Low switchover setpoint is bounded by the Hot Leg Switchover analyses of record.

The Hot Leg Switchover time analyses of record conservatively assume minimum temperatures to maximize mass when converting from volumetric to mass-based units for boration sources. The values assumed in the AORs are significantly lower than 70°F. Therefore, the HLSO AORs bound the change in minimum containment temperature.

The Westinghouse Hot Leg Switchover (HLSO) calculation methodology does not model the Containment Pressure – High setpoint, so this change has no impact on the HLSO AORs for BVPS-1 and -2.

The HLSO methodology does not model auxiliary feedwater, main feedwater, or containment spray. Therefore, the installation of the cavitating venturis in the AFW lines, the fast acting flow isolation valves in the Main Feedwater (MFW) lines, and the elimination of the Quench Spray flow cutback logic have no impact on the BVPS-1 HLSO AOR.

6.1.1.4 Post-LOCA Subcriticality and Long-Term Core Cooling

The LOCA related analysis employs primarily mass based calculations, while input data is often only available in volumetric units. The local temperature and pressure are used with the ASME steam tables to provide temperature/pressure dependent physical property values used to convert from volumetric to mass based units (density, enthalpy…). Because liquid is incompressible, and all of the relevant initial condition calculations assume sub-cooled or saturated water at containment pressure, the small change in pressure assumed for the containment has a negligible effect on those physical parameters used in the units conversion. Therefore, changing from sub-atmospheric to atmospheric containment pressure or reducing containment temperature has a negligible effect on the Post-LOCA Subcriticality and Long-Term Core Cooling calculations for BVPS-1 and -2.

The decrease in the RWST ECCS switchover setpoint that increases the water in the containment at the switchover point is bounded by the smaller volume used in the current calculation. Therefore, this change does not affect the Post-LOCA Subcriticality and Long-Term Core Cooling calculations for BVPS-1 and -2.

The BVPS-1 and -2 Post-LOCA Subcriticality and Long-Term Core Cooling calculations do not model the Containment Pressure – High setpoint and are not affected by this change.

The current Post-LOCA Subcriticality and Long-Term Core Cooling calculations do not model auxiliary feedwater, main feedwater or containment spray. Therefore, the installation of cavitating venturis in the AFW lines, the installation of the fast acting MFW isolation valves in the MFW lines, and elimination of the Quench Spray flow cutback logic have no impact on the BVPS-1 Post-LOCA Subcriticality and Long-Term Core Cooling AOR.

6.1.1.5 LOCA Hydraulic Forces

The Westinghouse LOCA hydraulic forces break model assumes a break force based on system pressure minus one half of the pressure at saturation temperature of the RCS fluid at the break location. Therefore, containment pressure changes have no impact on the LOCA Hydraulic Forces AOR calculations.

The additional changes in Section 6.1 do not impact this analysis because they are not modeled in the LOCA Hydraulic Forces calculations.

6.1.1.6 LOCA Transients Summary

Converting the containment design from sub-atmospheric to atmospheric has a negligible effect on SBLOCA, Hot Leg Switchover, and Post-LOCA Subcriticality and Long-Term Core Cooling (LTCC) calculations. The LOCA Hydraulic Forces are not affected by the atmospheric containment conversion, and the Large Break LOCA analyses of record performed for sub-atmospheric containment pressure bound operation at higher pressures. Therefore, converting the containment to atmospheric pressure is bounded by the existing LOCA analyses for BVPS-1 and -2.

Increasing the Containment Pressure-High setpoint has a negligible effect on the LBLOCA analyses. It does not affect the SBLOCA, HLSO, LOCA Hydraulic Forces, and Post-LOCA Subcriticality and LTCC calculations at BVPS-1 and -2.

The decrease in RWST Level – Low switchover setpoint is bounded by the current calculations for SBLOCA, HLSO, and Post-LOCA Subcriticality and LTCC at BVPS-1 and -2. The change has no impact on the LBLOCA or LOCA Hydraulic Forces analyses.

The decrease in the containment minimum temperature is bounded by the current calculations for HLSO at BVPS-1 and 2. The change has no impact on the LBLOCA, SBLOCA, LOCA Hydraulic Forces and Post-LOCA Subcriticality and LTCC calculations.

The elimination of the Quench Spray flow cutback logic on RWST Level-Low is bounded by the current BVPS-1 calculations for LBLOCA and SBLOCA. This change does not impact the current BVPS-1 LOCA Hydraulic Forces, HLSO, or Post-LOCA Subcriticality and LTCC calculations.

Furthermore, the addition of AFW cavitating venturis and feedwater isolation valves in BVPS-1 has a negligible impact to the SBLOCA and is not modeled in any of the other LOCA related analyses. BVPS-2 has no changes required to either the auxiliary feedwater or main feedwater systems.

6.1.2 Steam Generator Tube Rupture

An evaluation was performed relative to changing the containment pressure from sub-atmospheric to atmospheric. Containment pressure is not used as an input to the steam generator tube rupture (SGTR) analysis and, therefore the SGTR analysis is not impacted by conversion of containment to atmospheric. Similarly, the SGTR analysis does not model and is not impacted by the changes to ESFAS setpoints for containment pressure and RWST level.

Additionally, an evaluation was performed for SGTR relative to the impact of the installation of AFW cavitating venturis and feedwater flow isolation valves in BVPS-1. This evaluation confirmed that the installation of this equipment in BVPS-1 has no adverse impact on the results of the SGTR analysis for BVPS-1.

For BVPS-2, there are no changes to either the auxiliary feedwater or main feedwater systems. Therefore, there is no impact to the BVPS-2 SGTR analysis.

6.1.3 Non-LOCA Transients

An evaluation was performed relative to changing containment pressure from sub-atmospheric to atmospheric. The containment pressure is used as an input in the analysis for two non-LOCA events, i.e., Steamline Break and Feedline Break. In each of these analyses, increasing the containment back pressure or reducing containment temperature would either have no impact or provide a very slight benefit.

Additionally, an evaluation was performed for the non-LOCA events relative to the impact of the installation of AFW cavitating venturis in BVPS-1. The cavitating venturis are designed to limit AFW flow to a faulted (i.e., depressurized) steam generator and subsequently divert flow to the intact steam generators. When no steam generators are faulted, the cavitating venturis reduce AFW flow to all steam generators. The discussion below addresses each non-LOCA event that is adversely impacted by reductions in AFW flow.

At the current power level, the Loss of Offsite Power (LOOP) to the Station Auxiliaries (UFSAR Section 14.1.11), Loss of Normal Feedwater (LONF) (UFSAR Section 14.1.8), and Feedwater System Pipe Break (UFSAR Section 14.2.5.2) are mitigated by auxiliary feedwater. At the current power level, the LOOP and LONF functional requirement is 315 gpm auxiliary feedwater flow to 3 steam generators from one auxiliary feedwater pump. At the current power level, the functional requirement for feedline break mitigation is 300 gpm to the two intact steam generators (from one auxiliary feedwater pump) with auxiliary feedwater to the faulted steam generator isolated. An analysis has been performed to demonstrate that there is adequate margin in the motor and turbine-driven auxiliary feedwater pump curves to provide these functional requirements with the additional pressure drop from the installed cavitating verturis.

The installation of main feedwater isolation valves will have no adverse impact on any non-LOCA safety analyses. Current analyses assume feedwater isolation via the feedwater regulating valves. The introduction of an additional valve that can terminate feedwater will increase the reliability but does not adversely impact any analysis.

BVPS-2 has no changes required to either the auxiliary feedwater or main feedwater systems. Therefore, there is no impact to the BVPS-2 non-LOCA analyses.

6.1.4 Anticipated Transients without Scram

The potential impact of containment conversion on Anticipated Transients Without Scram (ATWS) is addressed in Section 7.14. The basis for the ATWS rule and the ATWS Mitigation System Actuation Circuitry (AMSAC) design discussed in Section 7.14 is Westinghouse generic reference analyses for 2-loop, 3-loop and 4-loop plant designs with different steam generator models. The reference analyses showed that the results for the 4-loop plant design are more limiting than those for the 3-loop plant design and demonstrated that the Westinghouse plant designs would satisfy applicable ATWS licensing requirements. Containment conversion does not impact these generic analyses or their applicability to BVPS-1 or BVPS-2. This conclusion includes consideration for the installation of AFW flow venturis and main feedwater flow isolation valves on BVPS-1 similar to what currently exist on BVPS-2.

6.1.5 Natural Circulation

The changes associated with containment conversion as listed in Section 6.1 have no adverse impact on the natural circulation flow and boron mixing capability of the Reactor Coolant System at BVPS-1 and -2. The natural circulation flow and boron mixing capability is not impacted by the increase in containment normal operating pressure consistent with an atmospheric containment design. The installation of AFW flow venturis and main feedwater flow isolation valves on BVPS-1 similar to what currently exist on BVPS-2 will not adversely impact the BVPS-1 natural circulation and boron mixing capability.

6.1.6 Initial Condition Uncertainties and Reactor Trip System/Engineered Safety Features Actuation System (RTS/ESFAS) Setpoints

The changes associated with containment conversion as listed in Section 6.1 have no adverse impact on the initial condition uncertainties (i.e., reactor power and RCS flow, temperature, and pressure uncertainties) used as input to the UFSAR accident analyses. Similarly, the changes associated with containment conversion will not impact the RTS/ESFAS setpoints except for the containment pressure and RWST level setpoints that have been revised as part of containment conversion. These changes to containment pressure and RWST level setpoints have been incorporated into the containment analysis performed at EPU conditions to support containment conversion and have been evaluated relative to their potential impact on Other Considerations (Chapter 6) and Programs (Chapter 7) at current power conditions.

The impact of containment conversion on plant instrumentation including RTS/ESFAS and containment pressure instrumentation is addressed in Section 6.4.

6.2 NSSS SYSTEMS AND COMPONENTS AND NUCLEAR FUEL

6.2.1 NSSS Control Systems

The changes associated with containment conversion as listed in Section 6.1 have no adverse impact on the design and operation of the NSSS control systems, including the Rod Control System, Steam Generator Level Control System, Pressurizer Pressure Control System, Pressurizer Level Control System, and Steam Dump Control System. The operability of the NSSS control systems is not adversely impacted by the increase in containment normal operating pressure consistent with an atmospheric containment design.

6.2.2 NSSS Fluid Systems

The changes associated with containment conversion as listed in Section 6.1 have no adverse impact on the design and operation of the NSSS systems, including the Reactor Coolant System (RCS), Chemical and Volume Control System (CVCS), Residual Heat Removal System (RHRS), and Safety Injection System (SIS).

6.2.3 NSSS Components

The changes associated with containment conversion as listed in Section 6.1 have no adverse impact on the design and operation of the NSSS components in the RCS, CVCS, RHRS, and SIS. The operability of the NSSS components located inside containment is not adversely impacted by the increase in containment normal operating pressure consistent with an atmospheric containment design. This includes the following NSSS components located inside containment:

- Reactor Vessel
- Steam Generators
- Reactor Coolant Pumps
- Control Rod Drive Mechanisms
- Loop Stop Isolation Valves
- NSSS Auxiliary Equipment

The impact of containment conversion on the Control Rod Drive Mechanism (CRDM) ventilation system operation is addressed in Section 6.3.4 and the impact of containment conversion on equipment qualification is addressed in Section 7.11. The impact of containment conversion on Motor Operated Valve (MOVs), Air-Operated Valve (AOVs), relief valves, and check valves is addressed in Sections 7.6, 7.7, 7.8, and 7.9, respectively.

6.2.4 Nuclear Fuel

The changes associated with containment conversion as listed in Section 6.1 have no adverse impact on the design or analyses for the nuclear fuel. Furthermore, they do not impact the nuclear fluence or the radiation-induced heat generation rates in the reactor internals associated with the nuclear fuel.

Since the radiological analysis for containment conversion was done at the uprated power, the core inventory of fission products is increased. In addition, RG 1.183 imposes applicability limits for the gap iodine fractions on the maximum linear heat rate in high burnup fuel. These limits will be reflected in future core designs.

6.3 SYSTEM IMPACT

6.3.1 Control Room Arrangement

Although the control boards are functionally and physically separate, BVPS-1 and BVPS-2 share a common control room. The control room areas of both units are open to each other and are therefore within the same pressure boundary. Both units share the emergency control room pressurization systems used during accidents. Neither the control room arrangement nor the emergency control room pressurization systems are affected by converting to an atmospheric containment.

6.3.2 Supplementary Leak Collection and Release System

The primary function of the Supplemental Leak Collection and Release System (SLCRS) is to ensure that radioactivity releases following a LOCA (i.e., due to leakage from the primary containment as well as components containing recirculating sump fluids outside containment) or a fuel handling accident, is collected and filtered for iodine and particulate removal prior to discharge to the atmosphere. The post accident site boundary and control room dose consequence analyses supporting atmospheric operation however do not credit filtration by the SLCRS.

With an atmospheric containment, the SLCRS will operate essentially the same as with a sub-atmospheric containment and will continue to meet its intended design conditions, without creating an unacceptable increase in post-accident radiation levels in adjacent areas due to increased filter shine resulting from long term containment leakage following a LOCA. Equipment Qualification zones and vital access routes in the vicinity of the SLCRS filters were reviewed to ensure continued compliance with the regulatory requirements of 10CFR50.49 and NUREG 0737, II.B.2. Therefore, converting to an atmospheric containment does not affect the design or operation of the SLCRS.

6.3.3 Containment Atmosphere Recirculation System

The containment atmosphere recirculation (CAR) systems for BVPS-1 and -2 operate as follows:

The CAR system is designed to maintain the bulk air temperature in the containment suitable for equipment operation during normal plant operation and consistent with Technical Specification containment air temperature requirements. The system is non-safety related.

The CAR system consists of three 50-percent capacity atmosphere recirculation unit coolers with air distribution ductwork. Each unit cooler consists of a motor driven fan and cooling coils.

Two of three atmosphere recirculation unit coolers function during normal operation. The cooling coils in each recirculation unit cooler assembly are served by a chilled water system during normal plant operation. Air is drawn by the fan over the respective unit's cooling coils and is discharged into, and distributed through, ductwork to the containment levels. Self-acting backdraft dampers are installed at the discharge of each fan to prevent reverse flow through an idle fan.

With conversion to an atmospheric containment, the system will operate essentially the same as a sub-atmospheric containment. System functional and performance requirements have not been significantly impacted. Key system parameters including fan performance requirements (i.e., effects of increased HP loading on design life of motors) have been reviewed to assure continued ability of the system to operate and to perform its design function under atmospheric conditions and at uprate power levels.

Based on system review, the CAR system is capable of operating under the pressure range conditions, temperature conditions, and at the increased heat loads associated with the EPU and conversion to atmospheric containment.

6.3.4 Control Rod Drive Mechanism Ventilation System

The control rod drive mechanism (CRDM) ventilation system is a forced air cooling system that provides for the removal of heat from the CRDM magnetic jack coils during normal power operation. Fans are used to draw containment ambient air down over the CRDMs and into a shroud ductwork. Each fan takes suction from the CRDM shroud, and then discharges the air through its respective cooling coil bank, and then back to the containment.

The BVPS-1 design contains three 50% capacity fans and cooling banks. Normally, the system has two fans in operation with the third fan available as a backup. The BVPS-2 design contains three cooling units, with each cooling unit containing two 33% capacity fans and one 33% capacity cooling coil bank. Normally, one fan per cooling unit runs continuously, while the other fan remains on standby.

The CRDM ventilation system is operated whenever the reactor coolant temperature is greater than 300°F / 350°F (BVPS-1 / BVPS-2), or if any CRDM is energized. Since the system has no accident mitigation function, it is not required to operate during accident conditions, and it is not designated as safety related.

The atmospheric containment conversion will not significantly affect either the containment ambient air temperature during normal power operation, or the amount of heat to be removed from the CRDMs. Therefore, there is no change in the system heat removal requirements, or the requirements of the cooling coils.

The higher density air will result in improved performance of the coolers, due to enhanced heat transfer rates. The CRDM ventilation system fans are capable of operating under both normal atmospheric pressure and sub-atmospheric pressure. Testing has indicated that adequate motor performance margin is available to allow operation at the proposed maximum normal containment pressure (14.2 psia).

Therefore, the increased containment pressure will not adversely affect the normal operation of the CRDM ventilation system.

6.3.5 Containment Atmospheric Filtration System

To enable access into the containment during normal operation, including both power operation and shutdown, a containment atmospheric filtration system is located within both the BVPS-1 and the BVPS-2 containment. These systems are designed to remove both airborne radioactive particulates and iodine gases. Each system consists of two 100% fans and two 100% filtration units, containing a charcoal adsorber and HEPA filters. These systems are typically idle, and are used at the discretion of the plant operator, depending upon plant conditions. These systems are not safety related.

The atmospheric containment conversion will not affect the reactor coolant leakage rates, and thus, will not affect the quantity of airborne radioactivity the filtration systems are to remove.

The containment filtration system fans are capable of operating under both normal atmospheric pressure and sub-atmospheric pressure. Adequate motor performance margin is available to allow operation at the maximum operating containment pressure.

Therefore, the increased containment pressure will not affect the operation of the containment filtration system.

6.3.6 Containment Vacuum System

The BVPS-1 and -2 Containment Vacuum Systems are each comprised of a steam jet air ejector with redundant vacuum pumps. The components are located outside containment and take a direct suction from the containment atmosphere. These components are non safety related; however, the piping penetrating the containment and the associated containment isolation valves are safety related. The safety related portion of the system isolates for the purpose of containment integrity during a design bases event.

The containment pressure change associated with conversion from sub-atmospheric to atmospheric containment is within the existing operating range capability of the piping, ejectors, vacuum pumps and containment isolation valves. The equipment is located outside of containment. Thus there will be no significant detrimental effect to the equipment or piping.

Currently, the containment steam jet air ejectors are used to evacuate a large amount of air and bring the containment internal pressure to a sub-atmospheric condition in a short period of time during a start up prior to plant operation. At this sub-atmospheric condition, the containment is maintained by the vacuum pumps and operates within analysis requirements and the containment air in-leakage rate can be checked.

After the conversion to an atmospheric containment, the steam jet air ejectors will continue to be used to bring the containment atmosphere to within the Technical Specification required containment pressure range. Operating procedures will be revised to reflect the revised operating range.

Currently, the containment vacuum pumps maintain the reactor containment internal pressure within the Technical Specification required range. After the conversion to an atmospheric containment, the containment vacuum pumps will continue to be used to maintain the containment internal pressure within the Technical Specification required range. The containment internal pressure will be maintained by turning the containment vacuum pumps on and off manually in the control room as is done currently. Operating procedures will be revised to reflect the revised operating range.

The containment vacuum pumps are originally sized to maintain containment internal pressure based on an expected load (e.g., containment in-leakage). The conversion to an atmospheric containment will reduce the in-leakage because Technical Specification required pressure range is being increased to pressures closer to atmospheric pressure. Thus the vacuum pumps capacity is expected to remain adequate after the containment conversion.

No change will be required to safety-related portions of the systems.

6.3.7 Containment Instrument Air System

The BVPS-1 and -2 Containment Instrument Air Systems supply air throughout each units containment building for instrumentation and control functions. The Containment Instrument Air Systems are not safety related. Each unit’s Containment Instrument Air System operates differently.

Both BVPS-1 and -2 have the air compressors located outside containment. BVPS-1 air compressors take suction from outside the containment. BVPS-2 air compressors take suction from inside the containment. Both are designed to operate at atmospheric conditions and will thus perform acceptably after containment conversion.

Since it takes suction from outside containment, the BVPS-1 air leakage tends to build up inside containment. The Motor Operated Vacuum Pumps are used to prevent containment pressure build up. Experience has shown the Motor Operated Vacuum Pumps are capable of maintaining the BVPS-1 containment well within the Technical Specification required containment pressure range. Conversion to Atmospheric Containment will increase the normal operating pressure inside the containment and thus very slightly reduce the leakage from the instrument air system into containment. This will slightly reduce the flow requirements imposed on the vacuum pumps. Thus the BVPS-1 Instrument Air System will continue to perform satisfactorily.

The BVPS-2 compressors are located outside containment but take suction and discharge to containment. Therefore, they make no net change in the containment air volume and will not impact containment pressure.

No changes are required for atmospheric containment operation, no maintenance changes are required, and no changes are required for operation procedures.

Setpoints have been evaluated and found acceptable without change.

Thus, conversion to atmospheric containment will not impact the operation of these compressors

6.3.8 Quench Spray System

The quench spray system is designed to provide cold water from the refueling water storage tank (RWST), chemically treat the water and spray the containment after a containment isolation Phase B (CIB) signal.

The quench spray system is made up of two separate parallel trains, each consisting of a quench spray pump discharging to a spray header located near the top of the reactor containment. Each train draws water independently from the refueling water storage tank. Each train operating alone is capable of delivering 100 percent of the required quench spray to containment.

Initially, the quench spray system runs alone. The RWST water is sprayed inside containment to remove heat and then accumulates in the containment sump. After a period of time when adequate recirculation spray pump net positive suction head (NPSH) is available, the recirculation spray system is started and the quench spray and recirculation systems run concurrently. The quench spray pumps are manually stopped when the RWST water level drops below a predetermined level to prevent cavitation.

With an atmospheric containment, the system will operate essentially the same as with a sub-atmospheric containment. System functional and performance requirements have not been modified with the exception of the proposed deletion of a cutback system at BVPS-1, an increase in the maximum spray temperature, and a higher CIB setpoint. The purpose of the cutback system was to reduce quench spray (QS) flow on a low level setpoint on the RWST to extend the spray duration for sub-atmospheric pressure control. With conversion to atmospheric containment, it is no longer necessary to return the containment to sub-atmospheric pressure within 1 hour after a LOCA. Thus quench spray cutback will be deleted. Key system parameters including pump performance requirements, NPSH, and effects of temperature changes on piping and components have been reviewed to assure continued successful operation of the system to perform its design function at both the current and uprated power levels, and at atmospheric conditions.

6.3.9 Recirculation Spray System

The recirculation spray system provides a means for long-term cooling utilizing the river water system (referred to as service water on BVPS-2) as the ultimate heat sink.

The recirculation spray system is designed to recirculate water from the containment sump through coolers to spray containment after a containment isolation Phase B signal and predetermined time delay. The time delay is to ensure adequate net positive suction head (NPSH) is available for the pumps and to avoid pump operating difficulties due to vortexing.

The recirculation spray system is made up of four system headers, each with an associated recirculation spray pump and recirculation spray cooler. Each BVPS-1 header feeds a 180 degree spray ring header located beneath the top of the reactor containment. Each header for BVPS-2 feeds a 360 degree spray ring shared with another pump. Two recirculation spray headers are required.

In order to provide additional NPSH for the recirculation spray (RS) pumps at BVPS-1, cold quench spray (QS) water is diverted from the QS pump discharge through orifices to the RS pump suctions for subcooling. Approximately 150 gpm is diverted to each inside RS pump, and approximately 300 gpm is diverted to each outside RS pump.

The water from the sump is recirculated through recirculation spray coolers where it is cooled by the service water system. The cooled water is then used to spray the containment and the cycle repeats itself for an extended period after the design basis accident.

No physical changes are being made to the system to support atmospheric containment or lower minimum containment temperature.

6.3.10 Chemical Addition System

The primary safety function of the Chemical Addition System is to provide for pH control of containment sprays and the containment sump in order to meet the assumptions used in the radiological dose analysis. The partition of iodine between the liquid and gas phases inside containment is strongly dependent on the alkalinity of the solution [reference NUREG/CR-5732 and Regulatory Guide (RG) 1.183].

Other safety functions include corrosion control and environment control. Experience has shown that maintaining the long-term pH of borated solutions in the specified range of 7.0 - 10.5 will help to inhibit initiation of stress corrosion cracking of austenitic stainless steel components used in the ECCS [Reference NUREG-0800, BTP MTEB 6-1, R2]. Controlling the long-term sump pH also ensures that class 1E components and other pH-sensitive elastomers located inside containment will remain qualified to operate within the post-LOCA environment.

The Chemical Addition System consists of a sodium hydroxide (NaOH) storage tank and associated chemical injection pumps and piping. The NaOH solution from the chemical addition tank is added to the quench spray water by a positive displacement pump. The system starts automatically on actuation of a CIB signal, and operates continuously until the RWST suction is terminated.

The system will perform its safety function in spite of a single failure of any active component, and is capable of elevating all water in the containment sump to a minimum pH above 7.0 well within 16 hours. The calculations were based on maximum boron concentrations and water volumes permitted by the Technical Specifications for the containment sump water sources, e.g., Refueling Water Storage Tank (RWST), the Reactor Coolant System (RCS), and the Safety Injection Accumulators while assuming minimum permissible NaOH tank volumes and concentrations and minimum chemical injection pump flow rate considering a failure of one pump.

Key system parameters including pump performance requirements, NPSH, and effects of temperature changes on piping and components have been reviewed to assure contained successful operation of the system to perform its design function. Refer to Chapter 4.0 for additional discussion on the LHSI pumps NPSH from the containment sump. The low head and high head safety injection pump available NPSH have been evaluated for the lower RWST ECCS switchover setpoint and are acceptable.

6.3.11 Emergency Core Cooling Systems

The Emergency Core Cooling System (ECCS) is designed to remove the stored and fission product decay heat from the reactor core following a Design Basis Accident (DBA) and is started automatically on receipt of a safety injection signal. The ECCS consists of the high head safety injection (HHSI)/charging pumps, the refueling water storage tank (RWST), low head safety injection (LHSI) pumps, and the safety injection (SI) accumulators. The ECCS operates in the injection phase and the recirculation phase. The injection phase provides borated water from the RWST to the reactor vessel. The recirculation phase provides long-term post-accident cooling by recirculating water from the containment sump. The ECCS components are designed such that adequate core cooling is provided in the injection phase (i.e., minimum of two accumulators, one HHSI/charging pump and one LHSI pump) and the recirculation phase (i.e., minimum of one HHSI/charging pump and one LHSI pump).

With an atmospheric containment and a lower minimum allowable containment temperature, the ECCS will operate essentially the same as with a sub-atmospheric containment. System functional and performance requirements have not been modified with the exception of an increase in the injection water temperature and a lower RWST level setpoint, which initiates realignment of the ECCS from the RWST to the containment sump.

The increase in injection water temperature results from the containment analysis and was demonstrated acceptable by analysis discussed in Section 6.1.1 of the report. The RWST level setpoint for realignment from the RWST to the sump was lowered to cause more of the cooler RWST water to be inserted into the containment before initiation of the recirculation phase.

Key system parameters including pump performance requirements, NPSH, and effects of temperature changes on piping and components have been reviewed to assure contained successful operation of the system to perform its design function. Refer to Chapter 4 for additional discussion on the LHSI pumps

NPSH from the containment sump. The low head and high head safety injection pump available NPSH have been evaluated for the lower RWST ECCS switchover setpoint and are acceptable.

6.3.12 Piping Systems

The piping systems (pipe stress, pipe supports, and equipment nozzle and supports) evaluated for BVPS-1 and -2 containment atmospheric conversion include recirculation spray, low head safety injection, high head safety injection, river water/ service water, auxiliary feedwater (BVPS-1) and main feedwater (BVPS-1). The piping systems were evaluated in accordance with parameters specified in Section 4.7.8 and Table 4-22.

The evaluations for the recirculation spray and the low and high head safety injection piping systems considered effects due to potential increased sump water temperatures for containment atmospheric conversion at the uprated conditions before and during recirculation spray, and during cold leg recirculation and hot leg recirculation. The evaluations performed have concluded that these piping systems, with no modifications, remain acceptable and will continue to satisfy design basis requirements in accordance with applicable design basis criteria.

The evaluation of the river water and service water systems considered the effects of potential increased temperatures at the outlet of the recirculation spray heat exchangers due to the revised containment analysis for the containment atmospheric conversion. The piping downstream of the recirculation spray heat exchangers was evaluated for the effects associated with increased sump water temperature. It has been determined that the piping and supports remain acceptable and will continue to satisfy design basis requirements in accordance with design basis criteria, when considering temperature effects resulting from containment atmospheric conversion.

The BVPS-1 auxiliary feedwater piping system was evaluated for the plant modification for the addition of inline cavitating venturis. The cavitating venturis, required for containment atmospheric conversion, will be added in all three of the 3-inch auxiliary feedwater lines that attach to the main feedwater piping in the main steam valve house. The auxiliary feedwater piping system, with modifications to install cavitating venturis, will remain acceptable and will continue to satisfy design basis requirements in accordance with applicable design basis criteria.

The BVPS-1 main feedwater piping system was evaluated for the plant modification for the addition of feedwater isolation valves. The feedwater isolation valves, required for containment atmospheric conversion, will be added in all three of the 16-inch main feedwater lines in the main steam valve house. The main feedwater piping system, with modifications, will remain acceptable and will continue to satisfy design basis requirements in accordance with applicable design basis criteria.

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

6.3.13 River Water System

The BVPS-1 River Water System (RWS) supplies river water to components during normal plant operation, plant cooldown, and refueling operations. The RWS also provides cooling to essential safeguards equipment in the event of a Design Basis Accident (DBA). The RWS has three pumps; two river water pumps are required to be operable per the Technical Specifications, and one pump is operating during normal plant operation. The third pump is a swing pump that may be substituted for either of the two other pumps. One pump is required for safe shutdown. On a Containment Isolation Phase B (CIB) signal, the non-safety-related portion of the RWS is isolated to divert flow to the four recirculation spray coolers while still providing flow to the other safety-related equipment.

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

6.3.14 Service Water System

The BVPS-2 Service Water System (SWS), acts as the ultimate heat sink for the plant. The system supplies river water to various Turbine Plant and Reactor Plant heat exchangers during normal plant operation, plant cooldown, and refueling operations. The SWS also provides cooling to essential safeguards equipment in the event of a Design Basis Accident (DBA). The SWS has three pumps; two service water pumps are required to be operable per the Technical Specifications, and two pumps are typically operating during normal operation. One pump is required for safe shutdown. The third pump is a swing pump that may be substituted for either of the two other pumps. On a Containment Isolation Phase B (CIB) signal, the non-safety-related portion of the SWS is isolated to divert flow to the four recirculation spray coolers while still providing flow to the other safety-related equipment.

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

6.3.15 Main Feedwater System

The feedwater system is designed to supply heated feedwater to the steam generators and to maintain the steam generator water level during steady state and transient operations. The steam generator water level is maintained by positioning the feedwater control valves. The major components in the main feedwater system are the feedwater isolation valves, feedwater control valves, and the feedwater pumps.

With an atmospheric containment the main feedwater system will operate essentially the same as with a sub-atmospheric containment. The containment analysis performed for atmospheric containment operating at an uprated power requires isolation of the BVPS-1 feedwater flow faster than the current design to prevent containment overpressurization following a MSLB accident. To accomplish this function, a new fast acting feedwater isolation valve will be added in each of three (3) lines supplying feedwater to the steam generators at BVPS-1. The new isolation valves are capable of closing in less than 10 seconds. This is similar to the existing arrangement at BVPS-2.

6.3.16 Auxiliary Feedwater System

The auxiliary feedwater system provides an emergency source of feedwater to the steam generators, as a backup to the normal feedwater system. It may also be used to supply feedwater to the three steam generators during startup, hot standby, cold shutdown, Station Blackout conditions, and Anticipated Trip Without Scram (ATWS). The system operates under the following conditions: loss of normal feedwater, loss of offsite power followed by a reactor trip, secondary side pipe rupture, or cooldown following a steam generator tube rupture.

The major components in the auxiliary feedwater system are two motor-driven auxiliary feedwater pumps, one turbine-driven auxiliary feedwater pump, and the primary plant demineralized water storage tank.

The three pumps take suction through individual suction lines from the primary demineralized water storage tank. Each motor-driven auxiliary feedwater pump delivers water to a separate auxiliary feedwater header, with the turbine-driven pump manually aligned to one of the two headers.

With an atmospheric containment and current licensed power level, the auxiliary feedwater system will operate essentially the same as with a sub-atmospheric containment. However, reduced auxiliary feedwater flow to the depressurized steam generator is required to prevent containment overpressurization following a MSLB accident. To accomplish this function, cavitating venturi flow elements will be installed in each of the auxiliary feedwater supply lines to each steam generator at BVPS-1. This is similar to the existing arrangement at BVPS-2.

The minimum flow requirements for the auxiliary feedwater system are dictated by other accident analyses described in the UFSAR. This has been analyzed and discussed in Section 6.1 and the evaluation concluded that the performance is acceptable. The evaluation of the system at the uprate power level will be provided in the EPU LAR.

6.4 INSTRUMENTATION IMPACTS

Normal operation with nominal containment pressure at atmospheric conditions has the potential to impact instrumentation scaling, actuation, and indication for pressure related instrumentation. Atmospheric containment operation will affect pressure instruments configured as “gauge pressure” measurements (i.e., with a reference leg/connection open to containment atmosphere, as a point of reference for the measured pressure parameter of interest). Based upon a review of instrumentation located inside Containment, impacted instrument channels have been evaluated as either; acceptable as is; to be replaced; or to be rescaled to accommodate operation at the new nominal atmospheric conditions.

Changes in normal minimum / maximum temperatures associated with this atmospheric containment conversion may also (to a lesser degree) impact temperature-related instrumentation scaling, actuation, and indication. Such impacts will be similarly adjusted through required scaling and/or procedure references, to delineate the new nominal operating conditions.

6.4.1 Reactor Trip System/Engineered Safety Features Actuation System (RTS/ESFAS)

The RTS/ESFAS setpoints for both BVPS-1 and -2, have been reviewed for potential impact as a result of this change from the current sub-atmospheric to atmospheric containment operation.

RTS/ESFAS signals from the Containment Pressure instrument channels provide Containment Phase A (CIA) and Phase B (CIB) isolation signals. These isolation signals have been evaluated at their corresponding higher setpoints, and it has been concluded that the intended safety functions are performed within acceptable limits as noted per Section 6.1.6.

Additionally, independent of the above noted containment pressure sensing/reference leg impact assessment, the ESFAS function for automatic transfer from injection to recirculation, using the Refueling Water Storage Tank Level – Low channels, will be rescaled to new setpoint values based upon acceptable updated analytical limits as noted in Section 6.1.6.

6.4.2 Containment Pressure

The BVPS-1 and -2 safety-related Containment Pressure transmitters and their associated reference leg connections are located outside of containment, and are generally not impacted by the conversion to atmospheric containment operation. Each transmitter remains connected to its respective containment sensing line, as originally designed/configured. Containment conversion requires that the instrument channel scaling and applicable setpoint functions (including CIA/CIB setpoints, noted in Section 6.4.1) will be rescaled to account for the increased containment nominal operating pressure.

BVPS-1 and -2 Containment Pressure transmitters provide Main Control Board/computer indication and alarm, as well as applicable CIA/CIB (High, Intermediate High-High and High-High Containment Pressure) protective functions. Wide Range Containment Pressure channels are used in the Emergency Operating Procedures.

For BVPS-1 and -2, two Narrow Range Containment Pressure channels are used for operator indication and alarm as well as for containment leakage determination. The transmitters in these channels will be replaced, based upon the need for a larger calibrated span for atmospheric containment operation. The channels will be rescaled commensurate with the new transmitter span. Although these instrument channels are not used for specific Technical Specification compliance, transmitter replacement will assure consistency of information available to the Operators.

6.5 TECHNICAL SPECIFICATION IMPACTS

The BVPS-1 and -2 Technical Specifications, Technical Specifications Bases and the Licensing Requirements Manual were reviewed to identify changes required as a result of the proposed new containment operating conditions, the revised containment integrity analysis, and plant modifications planned in support of these changes. The proposed changes to the Technical Specifications along with changes to the Technical Specification Bases and Licensing Requirements Manual are presented in Attachments A, B and C of License Amendment Requests 317 and 190.

7 PROGRAMS

The containment conversion has the potential to affect programs that are developed and implemented by station personnel to demonstrate that topical areas comply with various design and licensing requirements. The plant programs and/or issues listed in Table 7-1 were reviewed to determine the impact due to the containment conversion.

For the programs listed in Table 7-1, the controlling procedures and processes for the programs and key reference items within the procedures were reviewed. Program sponsors, implementing organization personnel and other cognizant individuals were interviewed for those issues and programs that would be impacted by the containment conversion. Based upon the review of this information, the extent of impact by the implementation of the containment conversion was determined for the various issues and programs.

The review process resulted in two groupings; (1) changes required and (2) changes not required. The results of the review are summarized in Table 7-1 and discussed below.

7.1 SIMULATOR

The BVPS specific simulator, which mimics the actual control room, is primarily used for training of operations personnel. In addition to the overall physical likeness between the actual control room and the simulator, computer systems provide simulator responses that are intended to match actual plant conditions for the simulation of accidents and transients, to the greatest extent possible. To ensure that the simulator accurately reflects the plant status, physical appearance (hardware) and simulation of plant response (software), changes resulting from the containment conversion must be effectively communicated.

The implementation of the containment conversion will also result in changes in plant operating characteristics (software changes). These changes will range from simple changes to plant responses to accidents and transients.

Changes (hardware and software) that affect the control room and the simulator will be implemented through plant approved change processes. Copies of these change processes are procedurally routed to the Training Department and the simulator personnel implement appropriate changes.

7.2 TRAINING

Conversion to atmospheric containment will require revision of numerous training materials and retraining of staff persons whose qualification depends on those training materials.

A review of training potentially impacted by conversion to atmospheric containment will be completed and persons whose qualifications depend on such training will be re-qualified before implementation of atmospheric containment.

7.3 STATION BLACKOUT (SBO)

The Nuclear Regulatory Commission (NRC) requires that each nuclear power plant be able to cope for a specified period and recover from a Station Blackout (SBO) event per SBO rule (10CFR50.63). SBO is defined as the complete loss of AC power to the essential and non-essential switchgear busses. BVPS has been evaluated against the requirements of the SBO rule using NUMARC 87-00. Per this evaluation, an Alternate AC (AAC) source is available within one hour of the onset of SBO. BVPS-1 utilizes on-site Emergency AC (EAC) Power from BVPS-2 as an AAC power source using the permanently installed 4 kV SBO cross-tie with manual operator action. Similarly, BVPS-2 utilizes on-site Emergency AC (EAC) Power from BVPS-1 as an AAC power source using the permanently installed 4 kV SBO cross-tie with manual operator action. The assumptions, systems and equipment credited for SBO coping were reviewed for an impact resulting from conversion of atmospheric containment. The SBO coping capability at the current licensed power level was found to be unaffected by conversion to atmospheric containment.

7.4 INSERVICE TESTING (IST) PROGRAM

The BVPS-1 and -2 programs for inservice testing requirements follow the guidelines of ASME Section XI and the positions in Attachment 1 of Generic Letter No. 89-04, ‘Guidance on Developing Acceptable Inservice Testing Programs’, including Supplement 1. However, 10CFR50.55a does provide relief from testing requirements when conformance to the code would cause unreasonable hardship, without a compensating increase in safety.

The BVPS IST program describes the required ASME Section XI testing for ASME Code Class 1, 2 and 3 pumps and valves. Included within this program are cold shutdown and refueling frequency justifications for quarterly testing requirements. The cold shutdown and refueling frequency justifications associated with equipment located within the containment are based in part on limited access to containment due to sub-atmospheric design. These justifications will be evaluated, considering the proposed change to an atmospheric containment, and the IST program updated, where appropriate, as part of the implementation of the containment conversion LAR.

7.5 INSERVICE INSPECTION (ISI) PROGRAM

The ISI program will be updated in conjunction with modifications implemented for atmospheric containment.

7.6 MOV PROGRAM

The existing Motor Operated Valve (MOV) administrative program controls the design basis, maintenance, inspection, testing, and trending of BVPS-1 and -2 motor operated valves. The containment conversion will not change any program controls, or existing licensing commitment.

The BVPS MOV program identifies the safety-related valves that are required to perform specific safety functions following postulated accidents. The program ensures these valves satisfy the requirements of NRC Generic Letters 89-10 and 96-05. As a result these valves are designated as “89-10” program valves. All safety-related valves have been evaluated, and formal exclusions document the valves not included within the “89-10” program. A sub-atmospheric containment was not used as a basis for any valve to be excluded from the “89-10” program.

The MOV torque and thrust requirements are based upon the maximum valve operating differential pressure (DP). A review of the calculations of each “89-10” MOV DP indicates the containment atmospheric pressure is an input to calculate DP for several valves, in the following manner:

- The maximum containment pressure used in the calculations corresponds to the containment design pressure of 45 psig, which is not changed by the containment conversion.
- The calculations also conservatively postulate a pipe break, or system discharge, so the downstream side of a valve would be subjected to atmospheric pressure. Thus, for valves with downstream piping inside containment, the containment conversion results in a decrease in the calculated valve DP.

Additionally, the program evaluates valves that must be closed for passive pipe leak downstream of the valve. As a result of the containment conversion, the containment pressure is slightly higher in containment at the time of switchover to cold leg recirculation and subsequently to hot leg recirculation. Therefore, the pumps (Low Head Safety Injection – Unit 1 and Recirculation Spray – Units 1 and 2) that take suction from the containment sump for recirculation conditions would have a slightly higher shutoff pressure. Therefore, for valves that may need to be closed for a passive leak, the valve DPs are slightly higher. These valve calculations were reviewed and the valve DPs were found to be acceptable.

Therefore, the MOV DP calculations performed for a sub-atmospheric containment remain conservative (yields higher valve DP) for the containment conversion.

Containment conversion affects upon environmental conditions is evaluated in Section 7.11 of this report. This evaluation determined the equipment remains qualified to perform its accident mitigation function.

The MOV program includes considerations for the loss of motor starting torque at elevated temperatures. The GL 89-10 Program valves have been evaluated for operational temperature effects for containment conversion conditions. This evaluation indicates the original analyses remain adequately conservative.

The change in the sub-atmospheric containment operating conditions will not adversely affect any MOV currently within the 89-10 program. The containment conversion will, however, affect the design bases of individual valves included within the MOV program:

- The modification to eliminate the quench spray cutback will remove two motor operated valves from the “89-10” program.
- The modification to add new feedwater isolation valves will remove three motor operated check valves from the GL-89-10 Program.

7.7 AOV PROGRAM

The ability of air operated valves to fully close and/or open within specific time constraints is influenced by several factors. These primarily include; the type and number of accessories in the valve air set, the size and configuration of the control/operating air piping, the amount of operator spring pre-compression, and the atmospheric pressure under which the air operator and control air must exhaust. Due to the number of variables, valve closing/opening time must be determined by test. The tests are typically documented in the valve manufacturing documentation. Any change in any one of the above mentioned variables will affect valve stroke time.

Initial valve testing conducted by valve manufacturers to demonstrate valve stroke time was typically performed under atmospheric pressure and normal ambient temperature conditions. Given the assumption of having to operate under initial sub-atmospheric conditions the test conditions are considered to be conservative. Valve operational testing is conducted after installation to verify acceptability of the as installed configuration. These tests also include verification of stroke time. In addition, valves subject to stroke time restrictions require periodic testing in the plant. These tests are also done under atmospheric conditions. The change to atmospheric containment would not significantly affect the AOV stroke time.

7.8 RELIEF VALVES

Relief valves located in BVPS-1 and -2 containments were reviewed. The review was made to determine if the change from sub-atmospheric conditions to the conditions of an atmospheric containment would cause any individual relief valve not to perform to the associated system requirements.

Relief valves function by measuring the differential pressure between two sources of pressure. There are two (2) possible relief valve arrangements:

1. The relief valve monitors and limits the differential pressure between two closed systems not exposed to the containment atmosphere. In this case, since both closed system pressures are independent of containment pressure, the relief valves effectiveness in protecting the system is not affected by changes in containment atmospheric conditions. Thus the valves will continue to function properly with no changes.

2. The relief valve monitors and limits the pressure differential between a closed system and the containment atmosphere and limits the pressure of the closed system. When the set point differential pressure driving the relief flow is not changed, there is no impact on the relief valve flow rate i.e., the pressure differential driving the relief flow remains the same. Thus the relief flow remains unchanged. In this case, the increased containment atmospheric pressure will be reflected in a corresponding increase in system pressure required to open the relief valve. The increased system pressures exerted after conversion to atmospheric containment were each evaluated. All were found to remain within the system requirements. Thus the valves will continue to function properly with no changes required.

Thus it was concluded that no relief valve changes are required for licensing or design basis as a result of changing to an atmospheric containment.

7.9 CHECK VALVES

The BVPS containment design utilizes several weight-loaded and spring-loaded check valves inside containment as containment penetration isolation valves. These weight and spring-loaded check valves are designed to require, in order to open, a differential pressure in the direction of the flow exceeding the expected (sub-atmospheric containment) post Design Basis Accident (DBA) differential pressure.

Testing is performed to ensure these weight and spring-loaded check valves remain closed when the differential pressure in the direction of flow is less than 1.2 psid, and opens when the differential pressure in the direction of flow is ≥ 1.2 psid, but less than 6.0 psid.

Since leakage into the containment through incoming lines with check valves inside containment caused by passive failures of such lines between the containment penetration and the outside isolation valve is prevented, this design is advantageous for a sub-atmospheric containment, after the post DBA containment atmospheric pressure has been reduced below ambient.

After the atmospheric containment conversion, the post DBA containment pressure will no longer be required to be maintained sub-atmospheric. Thus, atmospheric leakage into the containment is no longer a concern. Containment out-leakage will be a concern for the atmospheric containment. The weight and spring loaded check valves will perform the required safety function after containment conversion without modifications.

Proposed changes to the Technical Specification surveillance requirements are presented in Section 2 of Enclosure 1 of the License Amendment Requests 317 (BVPS-1) and 190 (BVPS-2).

7.10 CONTAINMENT INTEGRITY

7.10.1 Containment Leakage Rate Testing

Containment leakage rate testing is performed as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. Overall integrated leak rate testing (Type A) and individual penetration testing (Type B for components with resilient seals and Type C for containment isolation valves) are periodically performed to demonstrate acceptable leakage rates. The minimum test pressure for this leakage testing (P_a) is the peak containment pressure resulting from the design basis loss of coolant accident. Currently, BVPS-1 P_a = 40.0 psig and BVPS-2 P_a = 44.7 psig.

The peak containment pressures resulting from the design basis loss of coolant accident, re-calculated to support the atmospheric containment conversion, show BVPS-1 P_a = 43.3 psig and BVPS-2 P_a = 44.9 psig.

Since the BVPS-1 re-calculated P_a is higher than the current value an evaluation was performed to determine if additional or confirmatory testing at the higher test pressure is required.

The last BVPS-1 containment Type A test was conducted in 1993. This test was performed at a minimum containment pressure which does not bound the new BVPS-1 P_a. Thus, additional/confirmatory Type A testing is required. A Type A containment leak rate test will be performed on BVPS-1 at a pressure equal to or greater than the revised P_a prior to conversion to atmospheric containment.

BVPS-1 Type B testing of electrical penetrations is performed at a test pressure which bounds the new BVPS-1 P_a. Thus, additional/confirmatory testing of the electrical penetrations is not required.

Other BVPS-1 Type B tests, and all BVPS-1 Type C tests are conducted at a test pressure which is less than the revised P_a associated with atmospheric containment conversion.

All of the BVPS-1 Type B and Type C testing not already tested at or above the revised P_a will be performed at the higher P_a, prior to implementation of the atmospheric containment conversion.

Since the BVPS-2 re-calculated P_a is higher than the current value, an evaluation was performed to determine if additional or confirmatory testing at the higher test pressure is required.

The last BVPS-2 Type A containment integrated leak rate test was conducted in 1993. Although the P_a at that time was 44.7 psig, the test was actually performed at containment pressure which bounds the revised P_a associated with conversion to atmospheric containment (revised ACC P_a = 44.9 psig). Thus, additional/confirmatory Type A testing is not required for BVPS-2.

BVPS-2 Type B testing of electrical penetrations is performed at a test pressure which bounds the revised P_a associated with conversion to atmospheric containment (ACC P_a = 44.9 psig). Thus, those tests remain valid for atmospheric containment conversion and additional/confirmatory testing of the BVPS-2 electrical penetrations is not required.

Other BVPS-2 Type B tests, and all BVPS-2 Type C tests are conducted at a test pressure which bounds the revised P_a associated with conversion to atmospheric containment (ACC P_a = 44.9 psig). Thus, those tests remain valid for atmospheric containment conversion and additional/confirmatory testing of other Unit 2 BVPS-2 Type B tests, and BVPS-2 Type C tests is not required.

7.10.2 Other Penetration Matters

A review of the licensing documentation was performed to identify any reference to a sub-atmospheric containment for justification of containment penetrations. This review did not identify any such areas for BVPS-2. However, several letters in the BVPS-1 docketed correspondence between BVPS and the NRC did contain discussions in which a sub-atmospheric containment was used, in part, in the justification of some aspect of an individual containment penetration. The following discussions identify these penetrations, and provide an evaluation of the effects of the atmospheric containment conversion. These evaluations determined that containment conversion does not change any prior conclusions.

Containment Purge Supply/Exhaust and Vacuum Ejector Containment Penetrations

The containment isolations for these penetrations utilize resilient type seals. Past industry experience with this type of seal has shown a failure rate that normally warrants more frequent testing. Documentation (References 1 - 3) concluded that the additional testing at BVPS is not required due, in part, to the fact that the sub-atmospheric containment would cause only inleakage in the event of a damper seal failure.

The evaluation of the potential for increased containment leakage due to an atmospheric containment conversion has concluded that more frequent testing is still not warranted, due to the following considerations:

- Technical specifications requiring these dampers to be closed during modes 1-4 are not being changed by this LAR.
- When the dampers are opened during refueling or cold shutdown, damper leakage testing will still verify acceptable sealing prior to entering mode 4.
- BVPS testing experience with these dampers has not indicated the need for increased testing.
- Environmental changes are still within environmental qualifications, so increased seal degradation is not expected.

Hydrogen Recombiner Discharge Piping Containment Penetrations

The containment isolation configuration for the BVPS-1 hydrogen recombiner discharge piping (penetrations 87 and 88) does not conform to any configurations described in 10 CFR 50, Appendix A General Design Criteria (GDC) 56. Documentation (References 4-5) concluded that an exemption from GDC 56 for these penetrations is warranted, based in part on the rationale that a sub-atmospheric containment would most likely cause only inleakage in the event of a piping failure.

An evaluation of the potential for increased containment leakage due to containment conversion has concluded that the GDC 56 exemption is still warranted, based on the following considerations:

- The hydrogen recombiner discharge line containment isolation valves remain closed and locked when the Unit is in modes 1 – 4.
- Personnel access to allow manual isolation of the recombiner discharge line containment isolation valves is still maintained.

Recirculation Spray Heat Exchanger River Water Radiation Monitor Piping Containment Isolations

The containment isolation configuration for the BVPS-1 recirculation spray heat exchanger river water outlet radiation monitor sample lines (penetrations 83, 84, 85, and 86) does not conform to any configurations described in 10 CFR 50, Appendix A General Design Criteria (GDC) 57. An exemption from GDC 57 for these penetrations (References 6-9) is based in part on the rationale that a sub-atmospheric containment would minimize radiation releases during the time required for operator action to manually isolate a heat exchanger with a leak from the recirculation spray system into the river water system.

An evaluation of the potential for increased containment leakage due to an atmospheric containment has concluded that GDC 57 exemption is still acceptable, due to the following considerations:

- Currently, the recirculation spray heat exchangers shell (recirculation spray) side is maintained dry, and the tube-side (river water) is currently maintained in a chemical wet lay-up to minimize corrosion. The integrity of the heat exchanger is demonstrated by periodic testing.

- Personnel access to allow manual isolation of the radiation monitor sample inlet valves is still maintained.

- The increased radiation release through the radiation monitor sample lines due to an atmospheric containment is not significant.

References

1. NRC letter to Duquesne Light Company dated 7/81, Purge/Vent Tests.

2. Duquesne Light Company letter to NRC dated August 27, 1981, Butterfly Isolation Valve Leakage Integrity Tests.

3. NRC letter to Duquesne Light Company dated December 12, 1984, Technical Specifications for Leak Testing of Purge/Vent Valves at Beaver Valley, Unit 1.

4. Duquesne Light Company letter to NRC, “Additional Information Supporting TSCR 130.”

5. NRC letter to Duquesne Light Company dated June 22, 1988, “Issuance of Amendment.”

6. Duquesne Light Company letter to NRC dated January 11, 1990, “Application for Exemption from GDC-57.”

7. Duquesne Light Company letter to NRC dated April 29, 1991, “Addition Information for Exemption Request.”

8. NRC letter to Duquesne Light Company dated June 3, 1991, “Beaver Valley Unit 1 – Environmental Assessment Related to Exemption from Requirements of GDC 57.”

9. NRC letter to Duquesne Light Company dated June 26, 1991, “Exception from General Design Criterion 57 – Beaver Valley Power Station, Unit 1.”

7.11 EQUIPMENT QUALIFICATION REVIEW

The qualification of BVPS electrical equipment has been reviewed with respect to the change in temperature and pressure environmental conditions resulting from the conversion to atmospheric Containment.

The objective of the electrical equipment qualification is to assure that safety related electrical equipment would be capable of performing its design basis function if a design basis accident should occur at the end of its normal service life. Only accident response equipment that is exposed to the harsh environment resulting from the accident is subjected to the environmental qualification process. This includes equipment both inside containment and outside containment. This chapter addresses the impact on equipment qualification from the change from operating with a sub-atmospheric containment at approximately 9 psia to operating with a containment pressure of (12.8 to 14.2 psia).

The environmental qualification of the BVPS electrical equipment consists of testing and/or analysis in accordance with IEEE 323. This is done to demonstrate functionality under the environmental conditions during its service life followed by the environmental conditions during and after a design basis accident for the recovery time required by the response.

Degradation of the non-metallic materials occurs principally by chemical change to the materials that result in property changes that could impair the materials performing their intended safety related function. The principle environmental conditions that contribute to chemical change of non-metallics and/or to their impairment to perform are temperature, pressure, radiation, chemical exposure and humidity. Also included in equipment qualification, but as part of the design process, are the seismic and mechanical cycles effects. The impact of the atmospheric conversion on equipment qualification due to the change in the pressure, temperature, and radiation are evaluated in this Chapter.

The containment pressure and temperature results are based on the discussion provided in Chapter 4. Development of the radiation doses are based on the discussion provided in Chapter 5.

7.11.1 Equipment Inside Containment

Normal Operation

One environmental condition is the normal service temperature. The increased pressure in the Containment during normal operation and operation at the uprated power level will not change the normal operating temperature to which any equipment inside the containment is qualified. The containment cooling system will maintain containment temperature within Technical Specification limits of 105°F. Therefore, there is no change to the qualified life of equipment inside the Containment.

Pressure is another environmental condition for normal operation. The small pressure change to the Containment pressure during normal plant operation has no impact on the degradation of non-metallic components such as seals, diaphragms, gaskets, grommets and other parts of the equipment.

For the normal radiation dose inside Containment there is no increase associated with the conversion to atmospheric operation.

Design Bases Accidents

As discussed in Chapter 4, revised accident temperature and pressure profiles were developed for the review of the equipment qualification of BVPS. Both the Loss of Coolant (LOCA) and Main Steam Line Break (MSLB) accidents are considered.

The accident peak pressure, as re-evaluated, is enveloped by the accident pressure used for the equipment qualification. Therefore, there is no change to equipment qualification for the revised containment pressure for design bases accident transients inside the containment.

The revised accident analyses show that the MSLB events in the containment exceed the existing equipment qualification peak temperature profile by approximately 5°F for Unit 1 and 12°F for Unit 2. The revised analyses for the LOCA also exceed the existing temperature profile at the end of the LOCA transients. BVPS-1's long term LOCA temperature profile exceeds the equipment qualification profile at 11 hours by up to 10°F over 2.5 days. BVPS-2's LOCA temperature profile also exceeds the equipment qualification profile at 4 hours by up to 10°F over 5.8 days. Figures 7-1 and 7-2 provide the revised composite accident profiles and comparison between the existing equipment qualification temperature profiles and the new profiles.

The EQ equipment inside containment will be verified to demonstrate that the affected equipment is qualified for the new peak temperature, for the accident transient temperature conditions and for the long term Post Accident Operability Time (PAOT). Qualification to the revised accident transient conditions will be evaluated by verifying the equipment qualification tests envelop the revised composite accident. The PAOT will be evaluated using the Arrhenius methodology to demonstrate meeting the required PAOT with margin.

The accident radiation dose inside containment does not change as a result of the conversion to atmospheric containment operation. The methodology for calculating the environmental conditions inside containment is unchanged and continues to rely on TID-14844 source term. This subject is discussed in more detail in Section 5.3.1.

7.11.2 Equipment Outside Containment

Normal Operation

The normal service temperature for equipment outside the containment will not change because there are no significant changes in either heat sources or the operation of the ventilation system.

The change to atmospheric pressure inside the containment for normal operation has no impact on the environmental conditions outside the containment.

For the normal radiation dose outside containment there is no increase associated with the conversion to atmospheric operation.

Design Bases Accidents

The accident temperatures outside the containment are not impacted by the DBA inside containment.

The results of a review of the high energy line breaks outside the containment in safety related areas show that they do not change with the exception of the Main Steam Line Break for BVPS-1. Cavitating venturis will be installed in each of three BVPS-1 auxiliary Feedwater lines serving he steam generators. The addition of the cavitating venturis affects the environment associated with a MSLB outside the containment. The equipment affected by the MSLBs outside containment in the Main Steam Valve House, Service Building, and Turbine Building have been evaluated, and are not required to mitigate the effects of the accidents or have performed their function prior to exceeding their qualification.

As discussed in Chapter 5, the post-accident environmental doses outside containment increase in some locations as a consequence of containment conversion. This increase takes into consideration for the impact of long-term post-LOCA airborne radiation levels and the increased buildup of activity within ventilation filters. The safety related equipment that is credited for mitigating a LOCA long term and located in EQ zones outside containment have been reviewed, and it has been confirmed that they remain environmentally qualified for their intended post-accident safety function. The impact of atmospheric conversion and associated changes are discussed in Section 5.3.2.1.

7.11.3 Equipment Qualification Review Conclusions

The results of the equipment qualification review shows that the equipment remains qualified to perform its accident mitigation function in response to an accident while the plant is operating with an atmospheric containment.

7.12 PROBABILISTIC RISK ASSESSMENT (PRA)

The current Probabilistic Risk Assessment (PRA) for BVPS is based on a sub-atmospheric containment design. The containment will remain slightly sub-atmospheric after the containment conversion; however, the containment pressure is being increased. An engineering evaluation was performed to determine the potential impact on the PRA as a result of the containment conversion. This engineering evaluation is limited to the impact of the containment conversion on the contributors to Large Early Release Frequency (LERF) identified in the PRA. The PRA of record used in this engineering assessment was the BVPS-1 PRA Update Report, Issue 2, dated September 5, 2003 to perform an evaluation on both BVPS-1 and -2 (Reference 1).

Contributor	% of Total LERF	Total LERF
Large Containment Bypass Prior to Core Damage (Faulted SGTR)	91.8%	
Large Containment Bypass Prior to Core Damage (V-Sequence)	8.0%	
Containment Overpressurization due to Direct Containment Heating from HPME	0.2%	
Total	100%	9.99E-07 per year

Several items related to the containment conversion that may impact the containment performance modeled in the PRA were identified and assessed:

Containment Isolation – The sub-atmospheric containment design allows for the elimination of consideration of pre-existing containment isolation failures since they would be detected and remedied during normal operation of the plant based on the inability to maintain a containment vacuum. This "operational" diagnostic of containment isolation failures can still be credited after containment conversion.

Containment Baseline Pressure and Temperature – The sub-atmospheric design provides a pre-accident containment pressure and temperature baseline that was below that which would be present with a slightly sub-atmospheric design. The potential therefore exists that the peak containment pressures analyzed in the PRA could increase by a similar amount resulting in a higher peak containment pressure and a potential increase in the containment failure probabilities used in the PRA for some severe accident phenomena.

Containment Oxygen – The conversion to a slightly sub-atmospheric operation will result in additional oxygen being present in the containment during a severe accident. If any of the present PRA analyses related to potential containment failure due to hydrogen burns were oxygen limited, the conversion could impact the containment failure probability due to hydrogen burns.

Of the contributors to LERF, the containment bypass contributors (interfacing systems LOCA and induced SG tube rupture) are unaffected by the containment conversion since the containment plays no role in mitigating fission product releases for these sequences. The containment conditions (operation slightly sub-atmospheric vs. sub-atmospheric) play no role in determining the contribution to LERF from this failure mode.

Therefore, only the LERF contributions from direct containment heating and containment isolation failures need to be explicitly considered.

The LERF contribution from direct containment heating (DCH) is controlled by the containment pressure rise at reactor vessel failure due to heat transfer between molten core debris that is entrained in the vessel blowdown and the containment atmosphere. The peak containment pressure from DCH is established by adding the pressure rise due to DCH to the baseline containment pressure. Therefore, this LERF contributor would be impacted by the containment conversion. The current PRA assessment concludes that containment failure due to DCH only applies to the tails of the uncertainty distribution for very low probability sequences. Direct Containment Heating (DCH) calculations show a maximum containment pressure, due to DCH, 109 psia. This relates to an increase of containment rupture failure of less than 1.0%. Since DCH only contributes 0.2% to LERF, this results in an increase in LERF of less than 0.002% or less than 1E-7/yr.

The LERF contribution from containment isolation failures in the PRA is assessed to be negligible. The addition of consideration of pre-existing containment isolation failures (isolation failures that are undetected during operation and therefore exist at the time of the accident) would result in an increase in the overall contribution to LERF. The probability of pre-existing containment isolation failures, based on historical data is 1.78E-04 per reactor-year. The current core damage frequency at BVPS-1 is 2.34E-5 per reactor-year. Therefore, the increase in LERF due to pre-existing containment isolation failures is 4.2E-09 per reactor year. Since this increase is less than 1.0E-07, it is considered to be of low risk significance.

Conclusion

An evaluation was performed to determine the potential impact on the PRA as a result of changing from sub-atmospheric to atmospheric containment conditions during normal operation. This evaluation assessed the impact of containment conversion on the contributors to LERF identified in the PRA. The evaluation concludes that the increase in LERF would be less than 1%. The overall LERF from the BVPS is on the order of 1E-06 per year; therefore a 1% increase would be on the order of 1E-08 per year. This is well within the criteria established in Reg. Guide 1.174 for acceptably small risk increases.

Reference

1. Beaver Valley Power Station Unit 1, “Probabilistic Risk Assessment Update Report,” Issue 2, September 5, 2003.

7.13 HUMAN FACTORS

The accident at Three Mile Island (TMI) led to the issuance of NUREG-0700 "Guidelines for Control Room Design Reviews," and NUREG-0737, "Clarification of TMI Action Plan Requirements." BVPS conducted a detailed control room design review, which was reviewed and approved by the NRC staff.

Newly installed instruments, systems or components are required to support atmospheric conversion. In addition, instrument-monitoring ranges may be changed, and in some cases indicator scales may be changed. These changes will be processed in accordance with BVPS design change procedure, which includes human factors engineering design principles.

7.14 ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

The requirement to mitigate a transient without reactor scram is imposed in 10CFR50.62. The mitigation function is performed by the ATWS Mitigation System Actuation Circuitry (AMSAC). AMSAC provides a backup system diverse and independent from the existing Reactor Protection System to initiate a turbine trip and initiate auxiliary feedwater flow. This protects the Reactor Coolant System from overpressure in the event of an ATWS with coincident loss of main feedwater.

The only potential impact to AMSAC was the addition of the auxiliary feedwater cavitating venturis. The design of the cavitating venturis is such to ensure that minimum auxiliary feedwater flows for ATWS continue to be met.

7.15 FLOW ACCELERATED CORROSION (FAC) PROGRAM

FENOC has a long-term FAC Monitoring Program for BVPS-1 and -2 that consists of selected portions of single and two-phase high energy systems. The purpose of this program is to ensure that high-energy piping is adequately monitored to prevent failure. The program defines the criteria for selecting piping segments, selecting points at which to make thickness measurements, determining how frequently to make thickness measurements, selecting the methods used to make thickness measurements, and making replacement/repair decisions which are in the scope of the FAC Program. The FAC program uses select input data such as plant operating history data, water chemistry history, piping segment design conditions and component data.

Installation of the AFW cavitating venturis and fast acting FW isolation valves may impact the FAC program. The FAC program will be changed to accommodate installation of the AFW cavitating venturis and the fast acting FW isolation valves as part of the plant modification program used to make the changes.

7.16 FIRE PROTECTION

The fire protection system for BVPS-1 and -2 has been designed such that any single fire will not cause an unacceptable risk to public health and safety. A fire will not prevent the performance of necessary safe shutdown functions, and will not significantly increase the risk of radioactive release to the environment. The station is designed on the basis of minimizing the use of combustible materials and of the use of fire-resistant materials to the greatest extent possible.

Appendix R requires that the plant be capable of performing a safe shutdown in the event of a fire. BVPS-1 employs a dedicated auxiliary feedwater pump that is located in the turbine building. The addition of the BVPS-1 main feedwater isolation valves will be designed to accommodate the addition of auxiliary feedwater flow via the dedicated auxiliary feedwater pump in the event of a fire. The requirement that cold shutdown be achieved following a fire for instances when offsite power and the residual heat removal system are unavailable will remain unaffected.

For BVPS-2, the conversion to an atmospheric containment has no impact on Safe Shutdown Report for any postulated fire.

7.17 RIVER/SERVICE WATER SYSTEM CONTROL AND MONITORING (GL-89-13)

Heat exchangers cooled by raw water systems are subject to plugged tubes from debris, silting, and biological fouling, all of which degrade the capability of the heat exchanger to perform its intended function. Industry experiences with degradation of safety-related heat exchangers lead to the issuance of Generic Letter 89-13, and the requirement to establish a program to ensure raw water cooled, safety-related heat exchangers retain the capability to perform their required safety function. The GL 89-13 program applies to safety-related heat exchangers cooled by the River Water System (BVPS-1) and the Service Water System (BVPS-2).

The GL 89-13 program includes periodic flow and heat exchanger differential pressure monitoring, and periodic inspections and cleaning. The atmospheric containment conversion does not directly affect the GL 89-13 program, other than a small reduction in the River/Service Water flow requirements to the recirculation spray heat exchangers.

7.18 RESPIRATORY PROTECTION PROGRAM

The current sub-atmospheric containments are maintained at < 10.5 psia air partial pressure (TS 3.6.1.4), the BVPS Respiratory Protection Program currently requires oxygen enriched Self Contained Breathing Apparatus (SCBA) for entry into containment during normal operation to avoid possible oxygen deprivation.

Avoiding the need for respirators is considered desirable, since a respirator negatively impacts workers efficiency and human performance.

Considering both the recommendations of OSHA 29CFR1910.134 and ANSI Z88.2, the BVPS Respiratory Protection Program will be revised to allow containment entry without SCBA during normal operation (provided there is no radiological, chemical or other reason for SCBA). This change is supported by an evaluation consistent with ANSI Z88.2, which re-defined the minimum containment atmosphere for entry without requiring respirators as an oxygen partial pressure > 122 mm Hg, which is equivalent to 12.2 psia. Since the specified operating range will be greater than 12.2 psia, it results in the elimination of the SCBA requirement for containment entries during normal plant operating conditions.

This evaluation is consistent with NRC guidance provided in NUREG/CR-0041Rev 1 “Manual of Respiratory Protection Against Airborne Radio-Active Materials; January, 2001” which discusses the requirements for entry into sub-atmospheric containment’s in section 6.13 and refers to ANS Z88.2-1992.

Respirator Protection will continue to be used as appropriate for protection from other risks such as radiation and chemical hazards.

Table 7-1
Programs/Issues

Programs and Issues	Requires Update
Plant Simulator	YES
Training	YES
Station Blackout	NO
Inservice Test Program	YES
Inservice Inspection Program	YES
Motor-Operated Valve Program	YES
Air-Operated Valve Program	NO
Relief Valves	NO
Check Valves	YES
Containment Integrity	YES
Equipment Qualification Review	YES
Probabilistic Risk Assessment (PRA)	YES
Human Factors	YES
Anticipated Transients Without Scram (ATWS)	NO
Flow Accelerated Corrosion (FAC) Program	YES
Fire Protection	YES
River/Service Water System Control and Monitoring (GL 89-13)	NO
Respiratory Protection Program	YES

No - Programs not impacted by containment conversion or are bounded by existing analysis.

Yes - Programs impacted and changes required to address containment conversion.

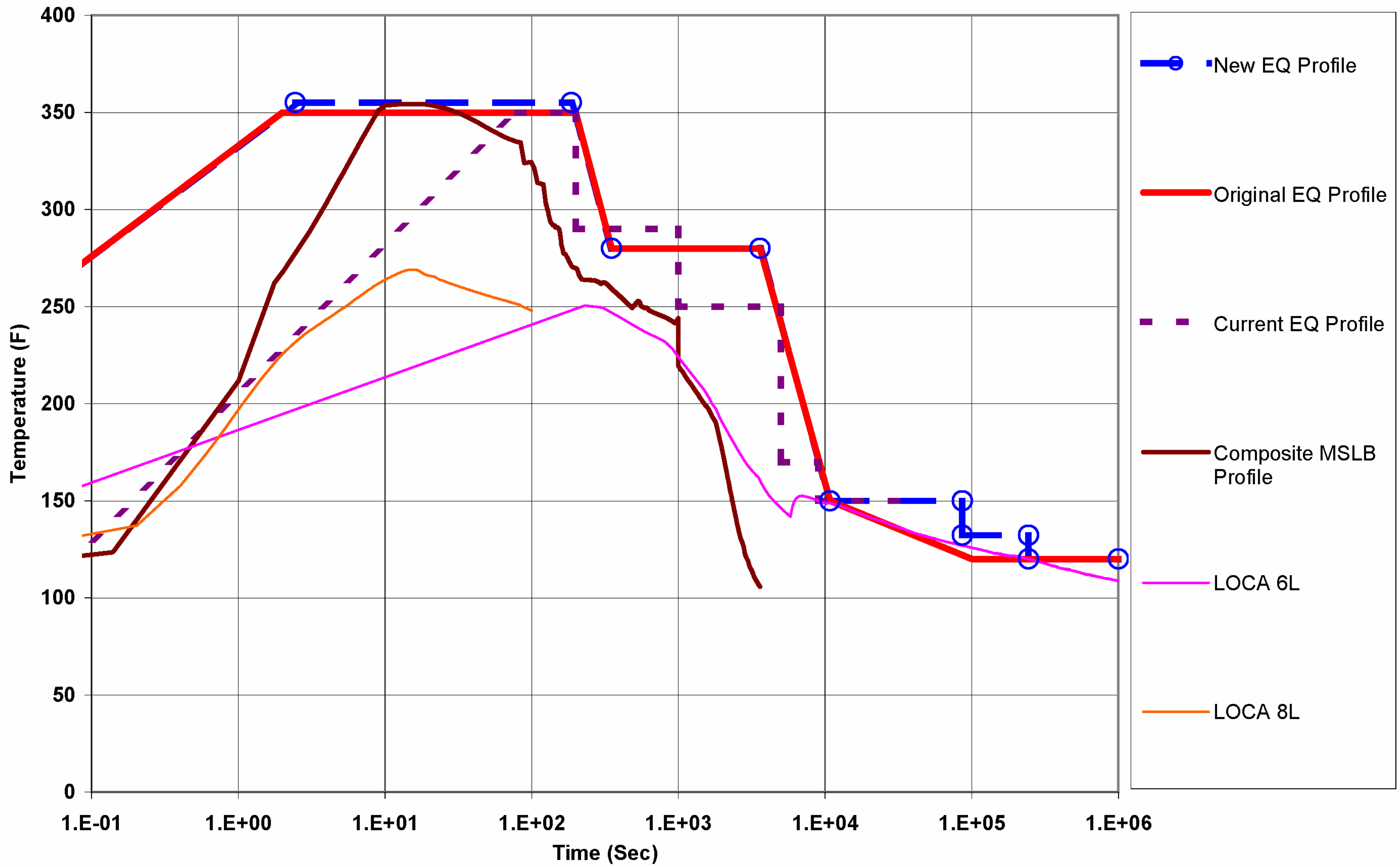


Figure 7-1
BVPS-1 EQ Temperature Profiles

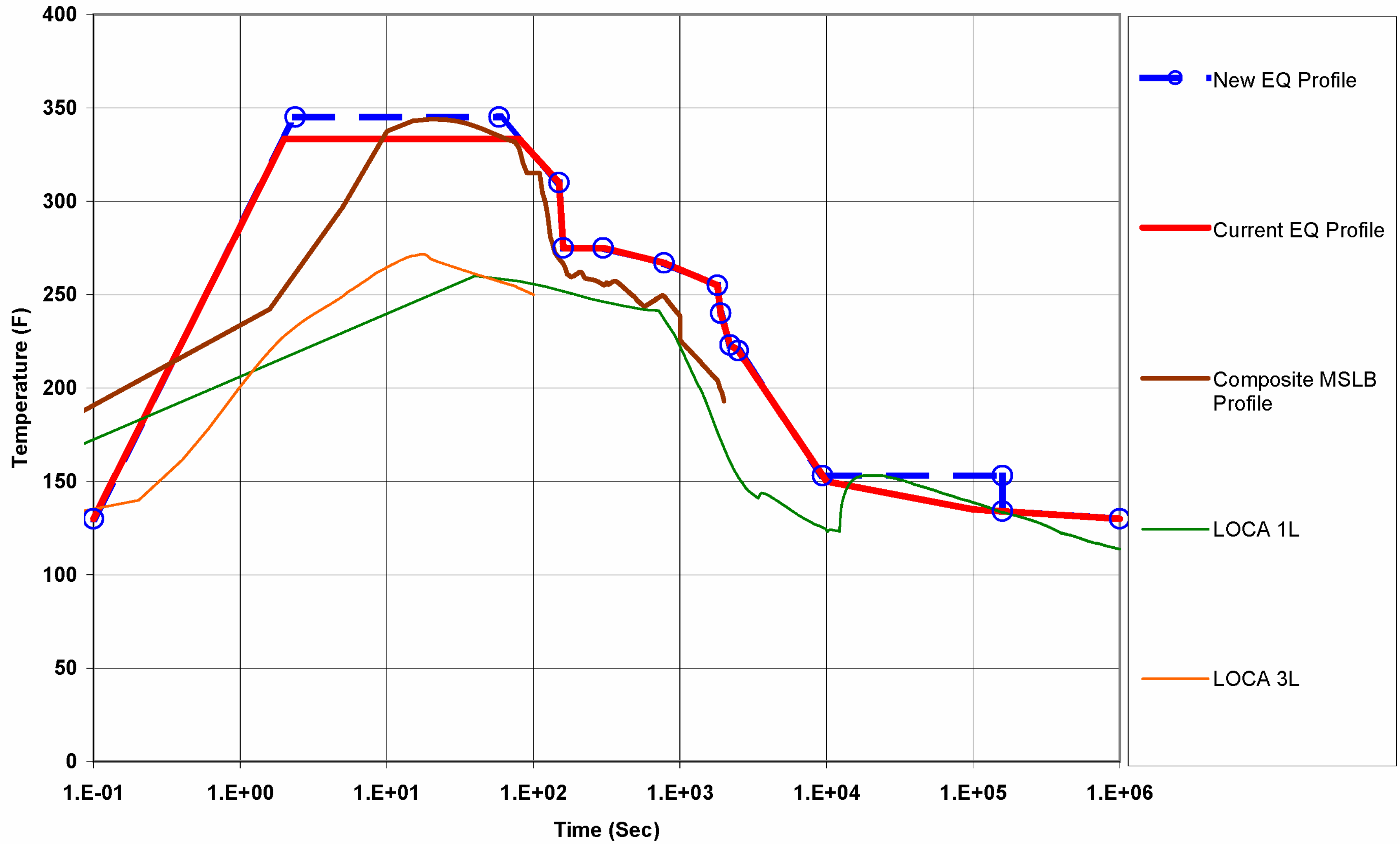


Figure 7-2
BVPS-2 EQ Temperature Profiles

8 ENVIRONMENTAL IMPACT

This chapter addresses the environmental impact of changing the normal operating pressure in the Containment Building from sub-atmospheric to atmospheric pressure. Also included is the environmental impact associated with eliminating the requirement of achieving sub-atmospheric conditions inside the containment within 1 hour following accidents that result in containment pressurization.

The NRC, in the Final Environmental Statement (FES) for BVPS-1 (Reference 2) and BVPS-2 (Reference 5), originally evaluated environmental issues associated with the issuance of the operating license for BVPS-1 and -2, respectively.

The FES documents (References 2 and 5) are considered historical licensing documents that will not be revised. However, the basis for the conclusion in the FES are maintained through adherance to the operating license and the National Pollutant Discharge Elimination System (NPDES) permit.

The BVPS-1 and -2 FES concluded that, after weighing the environmental, economic, technical, and other benefits against environmental costs and considering available alternatives, and subject to certain conditions, from the standpoint of environmental effects, the issuance of an operating license was an acceptable action. The FES conclusions are not impacted as a result of the containment conversion.

The environmental review conducted for the conversion of the containment from sub-atmospheric to atmospheric operation assessed the NPDES permit limits (Reference 1), the BVPS-1 Environmental Report - Operating License Stage (ER-OL) (Reference 3), as supplemented by the BVPS-2 ER-OL (Reference 6), and the BVPS-1 Final Environmental Statement (FES) (Reference 2), as supplemented by the BVPS-2 FES (Reference 5). This assessment included determining whether operating the containment at atmospheric conditions would cause the plant to exceed the NPDES permit limits or affect the conclusions reached in the FES.

NPDES Permit Impact

The NPDES Permit grants authorization to discharge liquid effluents from BVPS-1 and -2 to the receiving waters of the Ohio River and Peggs Run, in accordance with effluent limitations, monitoring requirements, and other specified conditions. The change to an atmospheric containment will have no impact on the liquid effluent discharges governed by the NPDES Permit.

Environmental Report – Operating License Stage and Final Environmental Statement Impacts

The Environmental Report – Operating License Stage (ER-OL), and the Final Environmental Statement related to BVPS-1 and BPVS-2 evaluated the impact of building and operating the plant against environmental parameters at the time of application.

The ER-OL, assessment includes the following environmental effects:

- Land Use Compatibility
- Water Use Compatibility

- Operation of Heat Dissipation Systems
- Radiological Impact from Routine Operation
- Chemical Discharges
- Sanitary Waste Discharges
- Switchyard and Transmission Lines
- Noise
- Resource Commitments
- Radiological Consequences of Accidents
- Transportation of Fuel and Radioactive Waste

The FES for BVPS-1 and -2 (References 2 and 5) provides the NRC's independent assessment of the environmental effects of station operation, using information from the licensee's ER-OL and other sources.

This review only addresses the plant environmental impact associated with the proposed change. Specifically, the change to containment operation from sub-atmospheric to atmospheric only impacts potential gaseous radiological effluent releases from the containment.

Environmental Review for Normal Operation

The Environmental Report (ER-OL) and the FES present the normal operation radiological releases and resultant doses, and they provide the basis for the original NRC acceptance.

Protection of the environment from radioactive effluents is assured by compliance with 10CFR50 Appendix I requirements. Radioactive liquid and solid waste effluent streams will not be affected by the change to an atmospheric containment. The change to an atmospheric containment will result in a small decrease in the radioactive gaseous effluents from the containment during normal operation. This reduction is due to increased decay time within the containment prior to release as a result of the decreased frequency of containment venting to maintain atmospheric vs. sub-atmospheric conditions.

In addition, review of the recent BVPS-1 and -2 Annual Radioactive Effluent Discharge Report, (Reference 4) demonstrates that the actual releases from the plants are a very small percentage of the 10CFR50 Appendix I limits.

Environmental Review for Postulated Accidents

The BVPS-1 Environmental Report (ER-OL) evaluated a variety of accidents, in a realistic manner, using guidance in the AEC document "Scope of Applicants' Environmental Reports with Respect to Transportation, Transmission Lines, and Accidents" issued on September 1, 1971. The NRC's independent assessment of the BVPS-1 ER-OL analyses, described in the FES, used the standard accident assumptions and guidance issued as a proposed amendment to Appendix D of 10CFR50 on December 1, 1971. Although the ER-OL methods and the FES methods do not exactly agree, the AEC concluded that the difference does not significantly affect overall environmental risk. It is noted that the Final Safety Analysis Report documented the dose consequences of the reference accidents using more stringent design basis assumptions than either the ER-OL or FES.

The BVPS-2 Environmental Report (ER-OL) evaluated a variety of accidents, in a realistic manner, using guidance of Regulatory Guide 4.2, Revision 2. The Final Safety Analysis Report documented the dose consequences of the reference accidents using more stringent design basis assumptions, and in accordance with the applicable regulatory guidance. The NRC's independent assessment of the BVPS-2 ER-OL and the FSAR analyses does not utilize realistic assumptions, but rather, estimates potential upper bound exposures for the purposes of implementing provisions of 10CFR50 and 10CFR100.

This review of the environmental impact of the atmospheric containment only assesses those accidents that are potentially impacted by the proposed change. It is noted that, in determining the impact of the change to containment operation at atmospheric pressure, analysis of the affected accidents uses the Alternative Source Term (AST) in accordance with 10CFR50.67.

The only analyses impacted by the change to an atmospheric containment are those accident analyses that take credit for post accident depressurization of containment to sub-atmospheric conditions. The revised accident analyses, utilizing the Alternative Source Term methodology outlined in NRC Regulatory Guide 1.183 and summarized in Chapter 5, demonstrate that accident radiological doses remain within the regulatory dose limits set by 10CFR50.67 and Regulatory Guide 1.183. The accidents that are not affected by the change remain within the regulatory dose limits set by 10CFR100.11 and NUREG 0800.

Protection of the environment and public is assured by compliance with 10CFR50.67 and 10CFR100.11.

References

1. NPDES Discharge Permit No. PA0025615, Commonwealth of Pennsylvania, Department of Environmental Protection, as amended effective June 1, 2003.

2. Final Environmental Statement related to the BVPS-1, U.S. Atomic Energy Commission, dated July 1973.

3. Environmental Report – Operating License Stage, BVPS-1, dated September 24, 1971, and Amendments 1 through 6.

4. FENOC letter to NRC, L-01-042, "BVPS-1 and -2 Annual Radioactive Effluent Release Report for 2000," dated 3/29/01.

5. NUREG-1094, "Final Environmental Statement related to the operation of BVPS-2," U.S. Nuclear Regulatory Commission, dated September 1985.

6. Environmental Report – Operating License Stage, BVPS-2, docketed on May 18, 1983.

9 DESCRIPTION OF THE MAAP-DBA CODE

9.1 INTRODUCTION AND PURPOSE

FENOC submitted a pre-application report via letter L-03-188, dated 11/24/03 that provided initial information using the MAAP-DBA version of the code to generate the limiting pressure cases for both BVPS-1 and -2. These cases showed that the results are within the currently licensed containment design pressure of 45 psig when operated at the EPU power level of 2900 MWt.

The MAAP-DBA code was developed to allow the calculation of containment response attributes for a spectrum of postulated LOCA and main steamline break sequences as part of design basis calculations for BVPS-1 and -2 containments. The containment assessments for design basis application are implemented in a manner consistent with the NRC guidance provided in the Standard Review Plan (Reference 1). This includes the use of the Tagami (Reference 2) and Uchida (Reference 3) heat transfer correlations for the quantification of the passive heat sink responses. The spectrum of containment response attributes to be quantified include the peak containment pressure, the short and long-term containment temperature, the containment liner temperature, the long-term sump water temperature, the available NPSH for ECCS and containment spray pumps, and the maximum service water outlet temperature for the containment heat removal heat exchanger. To address this set of containment response attributes for the spectrum of loss of coolant accident break sizes, both single node and multiple node containment models are used. The single node models apply for those design basis sequences and attributes that employ the Tagami and Uchida heat transfer correlations. For the multiple node applications, a heat and mass transfer analogy based on natural convection is used.

A single node model is used to calculate peak containment pressure and containment liner temperature as well as post accident containment global gas temperature profiles for equipment qualification. A multi-node model is used for NPSH and sump water temperature. This provides improved accountability of water hold up for NPSH and debris transport calculations.

9.2 OVERVIEW OF MAAP-DBA CONTAINMENT RESPONSE CALCULATIONAL APPROACH

Single node assessments will be used to quantify the containment peak pressure, gas temperature, and maximum liner temperature; while multiple node assessments will be used to quantify available NPSH and long-term sump temperature responses. MAAP-DBA is applied consistent with the Standard Review Plan (SRP) (Reference 1) and has the capability to model containment as a single node or as multiple nodes interconnected by flow junctions. This capability has been referred to as the generalized containment model (GCM) that replaced the fixed node and junction scheme included in previous versions of MAAP. The rates-of-exchange of mass and energy between multiple nodes are quantified as are the rates-of-exchange of mass and energy with active containment heat removal spray systems as well as passive heat sinks are also quantified. These are checked with mass and energy balances performed for each containment node.

The calculational procedure implemented in MAAP-DBA involves:

- an initialization step,

- the calculation of the auxiliary variables, such as the gas pressure and temperature, the water temperature, and various gas properties,
- the calculation of the rates-of-change of dynamic variables for each compartment by summing the rates-of-change of the physical phenomena and processes modeled,
- the saving of selected variables, and
- the output of selected variables to files for subsequent printing and plotting.

All compartments and heat sinks are initialized to the appropriate initial conditions, which are designated in the parameter file. Parameters that are initialized include the gas and water masses, the gas and water internal energy, compartment pressure and temperatures, and heat sink temperatures. The calculations of auxiliary variables update the gas pressure and temperature, the compartment water temperature, and various gas properties for nodes containing water, steam, and non-condensable by a call to MAAP-DBA subroutine PTCAL. In this manner, for each compartment the gas enthalpy, gas mole fraction, and water level are computed.

The calculation of the rates-of-change of the dynamic variables are compiled from various phenomenological models. These rates-of-change over the time step of interest are then integrated and then these changes in mass and energy are used to determine the new state and auxiliary variables. The calculation of the rates-of-change of mass and energy and heat sinks temperatures begins by setting the rates-of-change to zero. Then different processes are each computed, such as:

- engineered safeguards systems,
- passive metal and concrete heat sinks,
- flashing rates and pool evaporation,
- heat and mass transfer to spray droplets (computed using subroutine SPRAY),
- heat and mass transfer to suspended water droplets are calculated using subroutine RATES1, and
- flow rates between the primary system and the containment node, plus water and gas flow between multiple containment nodes.

The details of this functional scheme were discussed in the Beaver Valley Power Station Pre-Application Report (Reference 4). Section 9.4 of this chapter discusses the benchmarking of MAAP-DBA with important containment experiments and also with the GOTHIC code comparison.

9.3 MAAP-DBA CONTAINMENT MODELS

9.3.1 Single Node Containment Model

The design basis containment response calculations are implemented consistent with the intent of the Standard Review Plan. The containment peak pressure and temperature responses for large LOCA and main steamline breaks use the Tagami and Uchida heat transfer correlations to conservatively quantify the participation of the passive heat sinks. Implementation of these heat transfer correlations leads to the use of a single node containment model. Thus, the total containment volume and passive containment heat sinks are incorporated in a single node containment model that is applied for quantifying the peak pressure, peak gas temperature, and maximum containment liner temperature for the spectrum of main steamline break and large LOCA breaks. Furthermore, the containment liner temperature response is biased by using a multiplier of four (consistent with NUREG-0588 (Reference 5)) on the Tagami and Uchida heat transfer coefficients per the BVPS-1 and -2 current licensing basis.

The containment spray system impact varies by the accident sequence type. For the short-term large LOCA peak pressure and temperature responses, the time interval to the peak values is short and no containment spray system operation occurs prior to the peak condition. Conversely, main steamline break events, the peak conditions typically occur several hundred seconds following the postulated break. Thus, the quench spray system is activated and delivering spray to the containment atmosphere for most of the main steamline break sequence. A spray water droplet size of 1000 microns is modeled for heat and mass transfer. As a conservatism, no credit is taken for the recirculation sprays in determining the long-term temperature response for main steamline break calculations. For the main steamline breaks, the Uchida heat transfer correlation is used to describe the energy transfer rate to the passive containment heat sinks. The model used for the single node representation uses an 8% value (consistent with NUREG-0588 and NRC approved GOTHIC version 6.0a (Reference 6)) for the percent of condensate that is revaporized into the gas space when superheated conditions are experienced.

The model credits the impact of airborne water droplets originating in the blowdown jet following LOCA accidents. During and following the blowdown transients, airborne water acts as a heat sink for steam to minimize or eliminate superheat as well as to condense some of the steam mass. With the large surface-to-volume ratio, and high water density and specific heat compared to steam, a relatively small droplet water mass is sufficient to eliminate steam superheat from the containment atmosphere. During the blowdown the fraction of available liquid airborne is specified to be 10% of the non-flashed liquid blowdown. It is noted that the MAAP-DBA code using Tagami heat transfer correlation demonstrates a low sensitivity using this value. A droplet size for the airborne liquid mass of 100 microns is assigned. Following the end of the blowdown, the fraction of available liquid airborne is specified to be 0% which results in the remaining airborne water droplet mass decaying due to deposition and possibly spray operation.

9.3.2 Multiple Node Containment Model

The assessment of some of the long-term containment response attributes is conducted with a multiple node containment model. Specifically, the large break LOCA NPSH, the small break LOCA NPSH, and the small break LOCA sump water temperature attributes implement a multiple node model. The sump water level and temperature histories are key results to quantifying these specific attributes. Thus, the relative delivery rate and removal of water inventory from the containment sump and lower compartment influence the NPSH and sump temperature histories. Water hold-up from the break or spray injection sources in containment subcompartments directly influences the sump water level and temperature histories. Additionally, the distribution of containment sprays as they are collected on the operating deck floor can also influence these attributes. Thus, a multiple node containment configuration that identifies the elevations and sizes of junctions connecting the various containment regions is implemented for these evaluations.

The methods used to quantify the available NPSH for BVPS-1 and -2 are consistent with the current licensing basis. The BVPS-1 available NPSH method continues to include the overpressure credit, while the BVPS-2 available NPSH assessment does not credit overpressure.

The multiple node rate model uses natural convection heat transfer models for calculating the energy transfer to the containment heat sinks distributed through these multiple nodes. The natural convection heat transfer models are biased to minimize the calculated available NPSH. Like the single node model, the airborne fraction of LOCA blowdown is set to 10% with the droplet size set to 100 microns. Likewise, the spray system behavior is modeled in the same fashion as for the single node model.

The mass and energy releases from the primary system to containment for the spectrum of small and intermediate size LOCAs are generated using the MAAP-DBA code. The MAAP-DBA mass and energy release histories have been benchmarked with Westinghouse Small Break LOCA ECCS Evaluation Model (NOTRUMP). The mass and energy releases for the large break LOCA NPSH calculations are biased to yield the maximum sump water temperature by mixing the streams from the two sides of the guillotine break. As with the single node analyses, mass and energy release histories for each side of the break are quantified by applying the NRC approved Westinghouse methodologies (Reference 7).

9.4 BENCHMARKING OF MAAP-DBA MODEL

9.4.1 GOTHIC Comparison

Results from MAAP-DBA and GOTHIC version 6.0a have been compared for the peak containment pressure and temperatures associated with the limiting cases for both BVPS-1 and -2 containments. For the large LOCA comparisons, Case 8L was used for BVPS-1 and Case 3L was used for BVPS-2. For the main steamline break comparisons of peak pressure and peak gas temperature, Case 15M was used for BVPS-1 and Case 16M was used for BVPS-2. The large LOCA Cases 8L and 3L are for double-ended hot leg breaks from 100% reactor power. The Case 15M is a 1.4 ft^2 double-ended rupture from 30% power assuming the failure of the main steamline check valve. The Case 16M is a 1.069 ft^2 double-end rupture from 0% power with a main steam isolation valve failure and a failure of one train of quench spray. Table 9-2 summarizes the key input conditions used for this comparison.

The results of the comparison are provided in Table 9-3. Figures 9-1 through 9-8 provide a graphical comparison of the LOCA and MSLB pressure and temperature transients from each computer code. The use of a single node containment model and the Tagami and Uchida heat transfer correlations yield good agreement for the peak containment response attributes.

9.4.2 Separate Effects Experiments

Several separate effects experiments (listed in Table 9-4) were used for benchmarking models used in the MAAP-DBA containment code. In particular, the condensation heat transfer models and the containment spray heat removal models were benchmarked against available separate effects tests. The condensation experiments used included the University of Wisconsin flat plate experiments (Reference 8), the PHEBUS FPT0 experiments (Reference 9), and the Dehbi experiments (References 10 and 11). The containment spray heat removal benchmarks included the JAERI spray heat transfer tests (Reference 12) and the Kulic spray heat transfer experiments (Reference 12).

The Wisconsin flat plate steam condensation experiments in the presence of noncondensible gases (air) reported by Huhtiniemi, et al., are compared with the MAAP-DBA natural and forced convection heat transfer correlations by calculating the average heat transfer coefficients. MAAP-DBA calculated heat transfer coefficients for six different test conditions were compared with the data. The six tests included air-to-steam mass ratio of 0.29 to 3.5, bulk-to-surface temperature differentials of 20 to 60 degrees C, and flow velocities of 1 to 3 m/s.

In MAAP-DBA's validation, it was assumed that turbulence exists in the experiment due to mixing at the test section inlet such that the flat plate turbulent flow friction factor was used to calculate the forced convection heat transfer coefficient.

The comparisons of the MAAP-DBA calculated heat transfer coefficients against the Wisconsin flat plate data are shown in Table 9-5. As shown in the Table 9-5, the MAAP-DBA calculated heat transfer coefficients agree well with the data. At low air-to-steam mass ratios, the MAAP-DBA natural convection model is conservative.

The benchmark of the PHEBUS FPT0 test was performed to assess MAAP-DBA's condensation model. A comparison of the calculated containment vessel pressure against the measured pressure from PHEBUS FPT0 test is shown in Figure 9-9. Variations in the pressure history result from changes in the steam injection rate and the variation of the condenser surface temperature. MAAP-DBA over-predicted the peak pressure about 1.2 psi. In general, the calculated pressure transient follows the data very well for the most of the transient. Figure 9-10 shows the comparison between the measured and calculated condensation rates. The shape of the condensation rate transient follows the shape of the total pressure (i.e., the shape of the partial pressure of steam) and the calculated condensation rate agrees with the data. These results show that the natural convection heat and mass transfer model in MAAP-DBA is adequate to calculate the containment response under natural convection conditions.

The steam condensation experiments in the presence of noncondensible gases reported by Reference 10 are compared with the MAAP-DBA natural convection heat transfer correlation by calculating the average heat transfer coefficients. Three system pressures for the steam-air mixtures were studied with the steam mass fraction varied from very small concentration to values approaching 90% steam.

The correlation and the Dehbi data are compared in Figure 9-11. This figure illustrates the MAAP-DBA model prediction and the measured values (adjusted for curvature effects to represent a flat plate) for the different air mass fractions and pressures investigated in the experiments. As shown in this figure, the MAAP-DBA natural convection model reasonably predicts the measured heat transfer over the range of air mass fractions and containment pressures tested. An experimental uncertainty in the reported values of the "measured" heat transfer coefficients is ± 15%. The MAAP-DBA predicted heat transfer coefficient all are within this range of experimental uncertainty.

In the JAERI spray experiments, a large, tall vessel was pressurized with steam and subsequently cooled by an internal spray. The MAAP-DBA spray model was investigated for both single node and multiple vertical node models. In particular, Test PHS-1 was investigated since this provides an evaluation of both the pressurization and depressurization phases. The comparisons for the single node system are given in Figure 9-12 and for the multiple node system in Figure 9-13. As illustrated by these comparisons, the multiple node system provides a better representation of the overall transient. The single and multiple node model results both demonstrate the proper performance of the MAAP-DBA spray model for calculating heat removal and depressurization.

A set of spray droplet heat removal experiments were performed by Kulic (Reference 13) and used to benchmark the MAAP-DBA spray heat transfer model. Integral experiments were performed with water sprayed into a steam-air mixture contained in a large, closed vessel and the subsequent depressurization was recorded.

The MAAP-DBA calculated vessel depressurization was benchmarked against three Kulic tests, as shown in Figure 9-14. Test 1 illustrates the influence of heat losses from the test vessel when no spray flow was provided and confirms the initial and boundary conditions used for these benchmarks. Test 1A illustrates the influence of a single spray nozzle with a flow rate of 45 imperial gallons per minute (IGPM) at a temperature of 24°C (75°F). Test 1B illustrates the influence of five spray nozzles at a temperature of 24°C. The proper behavior of the MAAP-DBA spray model heat removal calculations is demonstrated by these single and multiple spray nozzle benchmarks.

9.4.3 Integral Effects Containment Experiments

The MAAP-DBA code has been benchmarked against integral effects containment experiments including several that have been used as International Standard Problems (ISPs). The containment design basis attributes of pressure and temperature will be quantified in these benchmarks and compared with the observations from the several integral experiments. Table 9-6 lists the integral effect experiments (IETs) used to benchmark the MAAP-DBA containment response.

These tests were used to demonstrate the prediction of the thermal-hydraulic response of large-scale multiple compartment containments. The comparisons show both the MAAP-DBA single and multiple node models used for the BVPS-1 and -2 containment analyses are conservative in predicting containment pressures and temperatures. The results from these benchmarks are provided below.

The decommissioned Heiss Dampf Reaktor (HDR) nuclear reactor containment was used to run large-scale design basis (LOCA and MSLB) experiments. Tests HDR-V44 (Reference 14) and HDR-T31.5 (Reference 15) are International Standard Problems that have been used to benchmark the performance of MAAP-DBA for predicting peak containment pressure and temperature responses. A two node model has been used, such that one node is for inside the HDR containment and the second node is for the annular gap that surrounds it. Consistent with the modeling used for the BVPS-1 and -2 analyses, in these benchmarks the Tagami heat transfer correlation is used and it is assumed that 10% of the non-flashed blowdown water becomes airborne as suspended water.

Comparison of the calculated pressure against the data for HDR-V44 is shown in Figure 9-15. MAAP-DBA over-predicts the peak pressure demonstrating that the MAAP-DBA Tagami heat transfer correlation and 10% airborne water is conservative. In terms of the gas temperature, the one node model provides an average gas temperature of the whole containment which is highly compartmentalized. The gas temperature at the vicinity of the break source is higher than the average gas temperature and the gas temperature at the lower elevation is lower than the average temperature. Figure 9-16 shows comparison of the calculated gas temperature against the measured gas temperature in the upper compartment which accounts about 43% of the total internal containment volume. The Tagami correlation with 1-node model over-predicts the gas temperature in the dome region.

Comparison of the calculated pressure against the data for HDR-T31.5 (Reference 14) is shown in Figure 9-17. MAAP-DBA also over-predicts the peak pressure for this test. Figure 9-18 compares the calculated gas temperature against the measured gas temperature in the upper compartment. The MAAP-DBA Tagami correlation and 10% airborne water with 1-node model also over-predicts the gas temperature in the dome region for these tests.

The NUPEC M-7-1 (Reference 15) test is an International Standard Problem that has been used to benchmark the performance of MAAP-DBA for predicting design basis containment responses with natural convection and the heat and mass transfer analog (HMTA) correlation. This test had two phases. The first phase was a “pre-heat” that simply discharged steam into the ¼-scale containment for three hours. This phase simulates a small LOCA sequence and is applicable to design basis events. The second phase initiated helium and steam injection plus containment spray operation to simulate a severe accident that experienced core damage and hydrogen generation. This phase of the M-7-1 test is not applicable to design basis events. The results for the pre-heat phase are reported here and used to benchmark MAAP-DBA’s containment responses.

The predicted containment pressure and temperature responses for this multiple node MAAP-DBA model are compared to the test data in Figure 9-19. A multiple node model is used for small break LOCA in the BVPS analysis. The code underestimates the heat transfer from the gases to the passive heat sinks. The containment pressure and gas temperatures calculated by MAAP-DBA are shown to over-predict containment pressure when compared to the test data.

The decommissioned Carolina Virginia Tubular Reactor (CVTR) containment was used to run large-scale tests that simulated MSLBs with and without containment spray actuation (Reference 16). The CVTR tests #3, #4, and #5 have been used to benchmark MAAP-DBA for predicting design basis containment responses with the Uchida heat transfer coefficient correlation. Test #3 did not include containment spray actuation, while the other two tests included sprays at two different flow rates. Measurements from the CVTR tests provide a set of large-scale containment response information to evaluate the dominant heat transfer processes associated with condensation, global natural circulation flows, counter-current natural circulation flows, and the influence of containment sprays.

Figure 9-20 illustrates the measured and calculated containment pressure and gas temperature histories for Test 3 using a single node MAAP-DBA model and demonstrates that a one node model overstates the containment pressurization, by more than 20%. Test 3 was selected because there are no sprays used, which is a more challenging test. (Once the containment sprays are turned on, the atmosphere is more homogenized and a single node gives an improved representation.) Furthermore, the calculation also overstates the containment temperature increase early in time and later in time, underestimates the temperature above the operating deck (TC-28) and overestimates the temperature below the operating deck (TC-5).

Figure 9-21 illustrates the measured and calculated containment pressure and gas temperature histories for Test 4 using a single node MAAP-DBA model. Once again, the single node model overstates the containment pressurization by more than 20%. Test 4 used containment sprays, and once they turned on, the containment pressurization and heat up were mitigated. A lower peak pressure was obtained than for Test 3, but significant margin is demonstrated. Likewise, the Test 4 calculation demonstrates a similar containment gas temperature response that overstates the temperature increase early in time and later in time, underestimates the temperature above the operating deck and overestimates it below the operating deck. Test 5 also included containment spray operation and its MAAP-DBA calculation demonstrated similar behavior as observed for Test 4.

The Battelle-Frankfurt Model Containment (BFMC) Test D-16 (Reference 18) is an International Standard Problem that has been used to benchmark the performance of MAAP-DBA for predicting design basis containment response with the Tagami heat transfer coefficient correlation and to provide a technical basis for the value used for the airborne water fraction of the non-flashed portion of LOCA blowdowns.

For containment integrity evaluations, the most important evaluations are the pressure histories in the compartment receiving the break discharge (break compartment) and the pressure imposed on the outer containment wall. Figure 9-22 compares the measured D-16 pressure in the break compartment with those calculated using MAAP-DBA. The corresponding calculated values for the pressure imposed on the model containment outer wall is given in Figure 9-23. The MAAP-DBA containment model calculated containment pressure conservatively bounds the measured values.

The calculated and measured temperatures in the break compartment are illustrated in Figure 9-24. Since a single node model is used for this benchmark, it yields the average containment temperature, and thus, under-predicts the temperature observed in the break node. Figure 9-25 compares the predicted and measured temperatures in a node that is remote from the break node. Here the predicted peak gas temperature bounds the measured value.

All saturated water blowdown experiments, such as the HDR tests and those performed at the Battelle-Frankfurt model containment show the aerosolization and airborne transport of water. Airborne water is manifested in at least two ways, the first is that the measured gas temperatures remain close to the saturation value corresponding to the measured containment pressure and the second is that airborne water is directly observed in the HDR tests. In the Battelle-Frankfurt Tests D15 (Reference 19) and D16, substantial amounts of water are transported to compartments away from the break room. At the end of test the measured water masses in these rooms are well in excess of those that could be accumulated in these locations due solely to condensation.

Of particular interest are the two Battelle-Frankfurt tests (D-15 and D-16) that are identified as Containment Analysis Standard Problems CASP1 and CASP2. The former was configured to blowdown from the top of the high pressure vessel whereas the latter experienced a two-phase discharge from the bottom of the vessel with about four times the water inventory that was used in D-15. While Test D-16 is more representative of large break LOCA conditions, both tests experienced two-phase blowdown rates. Furthermore, the configuration of the containment rooms differed between the tests with Test D-15 having the rooms aligned in a chain and the second experiment having two parallel flow paths from the break room to the outer containment compartment. At the completion of the tests, the water inventory collected in the various rooms was measured.

Tables 9-7 and 9-8 list the measured water inventories in each of the rooms, i.e., approximately 956 kg were measured in CASP1 (D15) with 3873 kg being found in CASP2 (D16). Both tables also list the concrete surface areas associated within each containment room and these are used to estimate the condensation that could have occurred in each of the rooms. Tables 9-9 and 9-10 show the estimation of the water phase remaining after flashing and evaporation were complete. These analyses show that 562.1 kg remained as water for CASP1 and 2404 kg in CASP2. Assuming that the steam mass condensed can be partitioned between the rooms in proportion to the fraction of the concrete heat sink surface area in each room, the mass condensed can be estimated and is listed in each table. Subtracting this estimated mass from the measured water mass approximates the water mass transported to these rooms as water. These tables illustrate that substantial water masses are transported to rooms removed from the break room. Summing the water masses transported as liquid outside of the break room and dividing by the total water mass remaining after flashing and evaporation, gives a value of 0.49 for CASP1 and 0.65 for CASP2. (Of these two, the second is by far the more meaningful since the test was conducted under large LOCA like conditions.) Consequently, these experiments undergoing a high pressure two-phase critical discharge clearly have a large fraction of water transported away from the break room. It is also noted that these experiments were performed with an impingement (baffle) plate immediately downstream of the break. These results demonstrate that the assumed airborne water fraction of 10% used in the BVPS-1 and -2 atmospheric containment assessment is a conservatively low value.

As expected, the single node containment models that applied the Tagami and Uchida heat transfer correlations over-predicted the peak containment pressures observed in this set of Integral Effects Experiments. The single node containment models also over-predicted the peak containment gas temperature, with the possible exception of the local break node temperature for LOCA simulations. For the measurements near the break room, the calculated values either exceed or are in close agreement with the reported values. Since a single node model yields the average containment gas temperature, it may under-predict the gas temperature after the peak occurs or in regions removed from the break location where non-condensables can accumulate for MSLB simulations.

9.5 MASS AND ENERGY RELEASES FOR SMALL BREAK LOCA

For the large break LOCA and main steamline breaks, the MAAP-DBA computer program used externally specified mass and energy release histories generated by NRC approved Westinghouse methodologies (References 7, 20 and 21) to quantify the containment response for several design basis response attributes, i.e., peak pressure, gas temperature, and containment liner temperature.

For the small break loss of coolant accidents, the MAAP-DBA computer code was used to generate the mass and energy release histories to quantify containment pressure as well as additional containment response attributes (ECCS sump water level and temperature and available NPSH for the recirculation spray and low head injection pumps). As discussed below, there is a precedent for using MAAP-DBA to quantify SBLOCA mass and energy releases. The release histories for a spectrum of break sizes in both the hot leg and cold leg regions of the primary systems are calculated. Benchmarks are performed against an alternate computer code that is also used to quantify primary system mass and energy releases for small breaks.

MAAP-DBA predicted mass and energy release histories have been benchmarked against two sets of release histories as generated by the NOTRUMP computer code (Reference 22) to confirm the acceptability of the MAAP-DBA predictions. In the first benchmark, an existing BVPS NOTRUMP application has been used. The NOTRUMP computer code was used to assess the BVPS reactor core (fuel temperature) response for a spectrum of small to medium break sizes. The mass and energy release histories that were quantified for 2, 3, 4, and 6 inch cold leg breaks by NOTRUMP have been used to benchmark the predicted mass and energy releases from the MAAP-DBA code. The 10CFR50 Appendix K requirements were incorporated in this NOTRUMP analysis. Thus, the decay heat curve based on the ANSI 1971 Decay Heat Standard with a 1.2 multiplier was used. Solely for the purpose of this comparison, the MAAP-DBA benchmark for these four cases also used this decay heat curve.

A second set of NOTRUMP mass and energy release histories were quantified specifically for benchmarking with MAAP-DBA . In this set of release histories both 2 inch diameter cold and hot leg breaks were calculated with NOTRUMP based on the ANSI 1979 Decay Heat Standard with a two standard deviation uncertainty. This is the decay heat curve that will be used in the BVPS-1 and -2 containment response quantification.

The integrated mass and energy release histories for each of these six cases are compared in Figures 9-26a through 9-31b. Good agreement is obtained for this spectrum of break sizes and locations. The trending of the releases is similar for MAAP-DBA and NOTRUMP. There are small instantaneous deviations for a given break size. The divergence in the initial release histories results from the more detailed models in NOTRUMP. However, the release histories tend to converge following the initial release interval and in some cases MAAP-DBA bounds NOTRUMP. When the MAAP-DBA integral mass release lags the NOTRUMP release, the rate of increase in sump water level may be slightly slower; this is conservative regarding available NPSH calculations. When the MAAP-DBA integral energy release exceeds the NOTRUMP release, the sump water temperature will be higher; this is conservative regarding available NPSH and thermal stress on affected piping. A large spectrum of break sizes (1 inch through 12 inch diameter) are analyzed to conservatively envelope the observed divergences such that the impact of the limited uncertainty in the MAAP-DBA mass and energy releases histories is bounded.

The reactor coolant system (RCS) model used by MAAP-DBA is the same as the MAAP 4.0 RCS model that has been used to calculate mass and energy (M&E) releases to the containment during small break LOCAs and submitted on other licensing dockets. The derivation of the M&E releases for BVPS is consistent with that used in MAAP 4.0 RCS model in the 1999 analysis conducted by FAI for D. C. Cook. The mass released from the MAAP 4.0 RCS model used in the D. C. Cook calculations were validated by comparison to the NOTRUMP code, considering a 2-in. diameter break. The comparison showed that the integrated break flow release to the containment calculated by the MAAP 4.0 RCS model was about 10% less than the comparable calculation by NOTRUMP. The NRC validated and accepted the D. C. Cook calculations (Reference 23) by comparisons to an audit calculation.

MAAP-DBA was also benchmarked with NOTRUMP in support of the AP600 project, considering a spectrum of small RCS hot leg break sizes ranging from 0.5-in. to 8.75-in. The results are documented in Reference 24. The benchmarking performed in this report shows good agreement between MAAP-DBA and NOTRUMP mass inventory calculations, except for one sequence that is specific to the AP600 design. The NRC accepted the use of MAAP4 in the AP600 PRA.

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Table 9-1
Recent Licensing Precedence

Application	Type	HTC	Airborne Water Fraction	Nodes	NRC Approval Status	Notes
W-312	Large Dry	Tagami/Uchida	100%	Single	Approved	GOTHIC Benchmarked to COPATTA
C-E	Large Dry	Tagami/Uchida	100%	Single	Approved	Benchmarked to CONTEMPT
W-412	Ice Condenser	Uchida	100%	Multi	Approved	Benchmarked to LOTIC
C-E	Large Dry	Tagami/Uchida	100%	Single	Approved	Benchmarked to CONTRANS
W-212	Large Dry	Tagami/Uchida	100%	Single	Approved	GOTHIC Benchmarked to COCO
AP600	Passive	Natural Conv.	5-100%	Multi	Approved	WGOTHIC

Table 9-2
Input Conditions for MAAP-DBA and GOTHIC Comparison

	GOTHIC Version 6.0a	MAAP-DBA
Nodes	Single	Single
Entrainment (Pools and Films)	Yes	No
Forced Convection	No	No
LOCA Airborne Water Droplet Fraction	10%	10%
Spray Droplet Diameter	1000 microns	1000 microns
LOCA Airborne Water Droplet Diameter	100 microns	100 microns
Re-vaporization	8%	8%
Initial Containment Pressure	14.2 psia	14.2 psia
LOCA: Heat Transfer (Short Term)	Tagami	Tagami
MSLB: Heat Transfer	Uchida with 8% revaporization	Uchida with 8% revaporization

Table 9-3
Summary of MAAP-DBA and GOTHIC Comparison

Unit	Sequence	Results Comparison							
		LOCA				MSLB			
		Pressure (psia)		Gas Temperature (°F)		Pressure (psia)		Gas Temperature (°F)	
		MAAP-DBA	GOTHIC	MAAP-DBA	GOTHIC	MAAP-DBA	GOTHIC	MAAP-DBA	GOTHIC
1	Case 8L	57.57	57.41	267.4	266.3	—	—	—	—
2	Case 3L	58.99	58.29	269.7	268.2	—	—	—	—
1	Case 15M	—	—	—	—	56.8	57.8	342.6	341.3
2	Case 16M	—	—	—	—	51.5	52.9	327.1	329.8

Table 9-4
Separate Effects Tests Used for MAAP-DBA Containment Response Benchmark

Benchmark	Test	Application
1.	U. of Wisconsin Flat Plate (Reference 9)	Condensation heat transfer (HMTA with forced convection used for multiple node models)
2.	PHEBUS FPT0 (Reference 10)	Condensation with non-condensables present
3.	Dehbi (References 11 and 12)	Condensation with non-condensables present
4.	JAERI PHS-1 (Reference 13)	Spray heat removal
5.	Spray Droplet Heat Transfer (Kulic) (Reference 14)	Spray droplet heat removal

Table 9-5
Comparison of MAAP-DBA Average Condensation Heat Transfer Coefficients for the Wisconsin Square Channel Experiments

Case #	T_{mix}, °C	T_w, °C	m_{air} / m_{steam}	V, m/s	h_{exp}*	$h_{exp\ (max,\ min)}$*	MAAP-DBA*/**
1	70	30	3.58	1	111.1	(122.2, 99.99)	113.9
2	70	30	3.58	3	213.9	(235.3, 192.5)	235.4
3	80	30	1.808	1	163.9	(180.3, 147.5)	165.2
4	80	30	1.808	3	305.6	(336.2, 275.0)	310
5	90	30	0.706	1	255.5	(281.1, 229.95)	256.3
6	95	45	0.31	1	546.	(600.6, 491.4)	402.9

* Heat transfer coefficient in w/m^2/K.

** MAAP-DBA uses the maximum of the natural and forced convection values. At 1 m/s, the code is using the natural convection value.

Table 9-6
Integral Effects Tests Used for MAAP-DBA Containment Response Benchmark

Benchmark	Test	Application
1.	HDR-V44	Large loss of coolant accident (LOCA)
2.	HDR-T31.5	Large LOCA
3.	NUPEC M-7-1	Small LOCA
4.	CVTR #3	Main steamline break without containment spray
5.	CVTR #4, #5	Main steamline break with spray actuation
6.	BFMC D-16	Large LOCA

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

Table 9-7
CASP1 Fraction of the Blowdown Mass Collected as Water in the Node

Node Number*	Volume (m^3)	t = 600 s			t = 3840 s			Concrete Surface Area (m^2)	Fraction of Concrete Area
		Measured Water Level (cm)	Water Mass Measured (kg)	Fraction of the Total Water Mass	Measured Water Level (cm)	Water Mass Measured (kg)	Fraction of the Total Water Mass		
R6	41.26	8.0	315	0.432	9.1	400	0.418	90.1	0.088
R8	40.53	5.2	175	0.240	6.1	245	0.256	91.7	0.090
R7	46.4	0	21	0.029	0	21	0.022	76.6	0.075
R4	12.2	6.3	75	0.102	7.1	85	0.089	38.6	0.033
R5	41.05	0	18	0.025	0.2	20	0.021	76.1	0.075
R9 (R9 Annulus) (R9 Center)	450 (300) (150)	 1.0 2.0	125 100 25	0.171 0.137 0.034	 1.8 2.5	185 150 35	0.194 0.157 0.037	 416.6 229.2	 0.409 0.225
TOTAL	625		729	0.999		956	1.000	1018.9	1.000

* Listed in the order of the flow path from the break discharge room to the outer room.

Table 9-8
CASP2 Fraction of the Blowdown Mass Collected as Water in the Node

		t = 840 s			t = 6480 s				
Node Number*	**Volume (m^3)**	**Measured Water Level (cm)**	**Water Mass Measured (kg)**	**Fraction of the Total Water Mass**	**Measured Water Level (cm)**	**Water Mass Measured (kg)**	**Fraction of the Total Water Mass**	**Concrete Surface Area (m^2)**	**Fraction of Concrete Area**
R4 (discharges into R5 and R7)	13.66	76.2	903	0.261	76.7	909	0.235	38.6	0.038
R5 (discharges into R9)	41.05	9.7	630	0.182	11.1	721	0.186	76.1	0.075
R7 (discharges into R8)	40.40	3.2	102	0.029	6.5	207	0.053	76.6	0.075
R8 (discharges into R9)	40.53	9.0	477	0.138	9.5	504	0.130	91.7	0.090
R9 (discharges into R6)	465.0								
(R9 Annulus)	300	6.3	967	0.280	7.2	1107	0.286	416.6	0.409
(R9 Center)	165	5.3	258	0.075	5.7	277	0.072	229.2	0.225
R6	41.26	4.7	122	0.035	5.7	148	0.038	90.1	0.086
TOTAL	641.9		3459	1.000		3873	1.000	1018.9	1.000

* Listed in the two parallel flow paths from the break discharge room (R4) to the outer room (R9) and eventually to R6.

Table 9-9
Interpretation of Water Mass Distribution for CASP-1

Initial water temperature = 286°C

Initial water enthalpy: $h_o = 1.267 \times 10^6$ J/kg

Two-phase mixture quality after blowdown: $x = \dfrac{h_o - h_f}{h_{fg}}$

Assume saturation at 50°C: x = 0.444

Steam in the atmosphere at 50°C ~ 52 kg

Total mass considered at 3840 secs = 956 + 52 = 1008 kg

Mass formed as steam due to blowdown = 0.444 x 1008 kg = 447.6 kg

Mass remaining as water after blowdown = 1008 – 447.6 = 560.4 kg

Estimated mass of water collected as condensed steam
$m_{w,con}$ = Fraction of Concrete Surface Area x 447.6 kg

Average airborne density of water is uniformly distributed throughout

$$\bar{\rho}_o = \frac{591 \ \text{kg}}{625 \ \text{m}^3} = 0.95 \ \text{kg/m}^3$$

Compartment (Room)	Water Mass Collected (kg)	Estimated Mass Condensed (kg)	Estimated Mass Transport as Water (kg)	$\bar{\rho} = \dfrac{m_v}{\text{Room Vol}}$ Airborne Density of Water	Fraction of Total Airborne $\bar{\rho}/\rho_o$
R6 (break room)	400	39.4	360.6		
R8	245	40.3	204.7	5.0	> 1.00
R7	21	33.6	0	0	
R7 and R8*	266	73.9	192.1	2.4	> 1.00
R4	85	14.8	70.2	5.8	> 1.00
R5	20	20**	0	0	
R9 Annulus	150	150**	0	0	
R9 Center	35	35**	0	0	
Total Water Mass Outside of R6	956	333.1	622.9		

*Provided for reference.

**Cannot be larger than the measured mass.

Fraction of the depressurized water jet measured outside of the break discharge room

$$= \frac{622.9 - 360.6}{560.4} = 0.47\,.$$

Table 9-10
Interpretation of Water Mass Distribution for CASP-2

Initial water temperature = 260°C

Initial water enthalpy: $h_o = 1.134 \times 10^6$ J/kg

Two-phase mixture quality after blowdown: $x = \dfrac{h_o - h_f}{h_{fg}}$

Assumed saturated conditions at 50°C: x = 0.388

Steam in atmosphere at 50°C ~ 55 kg

Total mass considered at 6480 secs = 3873 + 55 = 3928 kg

Mass formed as steam due to blowdown = 0.388 x 3928 = 1524 kg

Mass remaining as water after blowdown = 3928 – 1524 = 2404 kg

Estimated mass of water collected as condensed steam for a given compartment (room)
m_w,con = Fraction of Concrete Surface Area x 1524 kg

Average airborne density if water is uniformly distributed throughout the model containment
$\overline{\rho}_o$ = 2404/641.9 = 3.75 kg/m^3

Compartment (Room)	Water Mass Collected (kg)	Estimated Mass Condensed (kg)	Estimated Mass Transport as Water (kg)	$\overline{\rho} = \dfrac{m_v}{\text{Room Vol}}$ Airborne Density of Water	Fraction of Total Airborne $\dfrac{\overline{\rho}}{\rho_o}$
R4 (break room)	909	57.9	851.1		
R5	721	114.3	606.7	14.8	> 1.00
R7	207	114.3	92.7	2.30	0.61
R8	504	137.2	366.8	9.05	> 1.00
R9 Annulus	1107	623.3	483.7	1.61	0.43
R9 Center	277	342.9	0	0	0
R6	148	134.1	13.9	0.03	0.008
Total Water Mass Outside of R4	3873		2414.9		

Fraction of the depressurized water jet measured outside of the break discharge room
$= \dfrac{2414.9 - 851.1}{2404.0} = 0.65$.

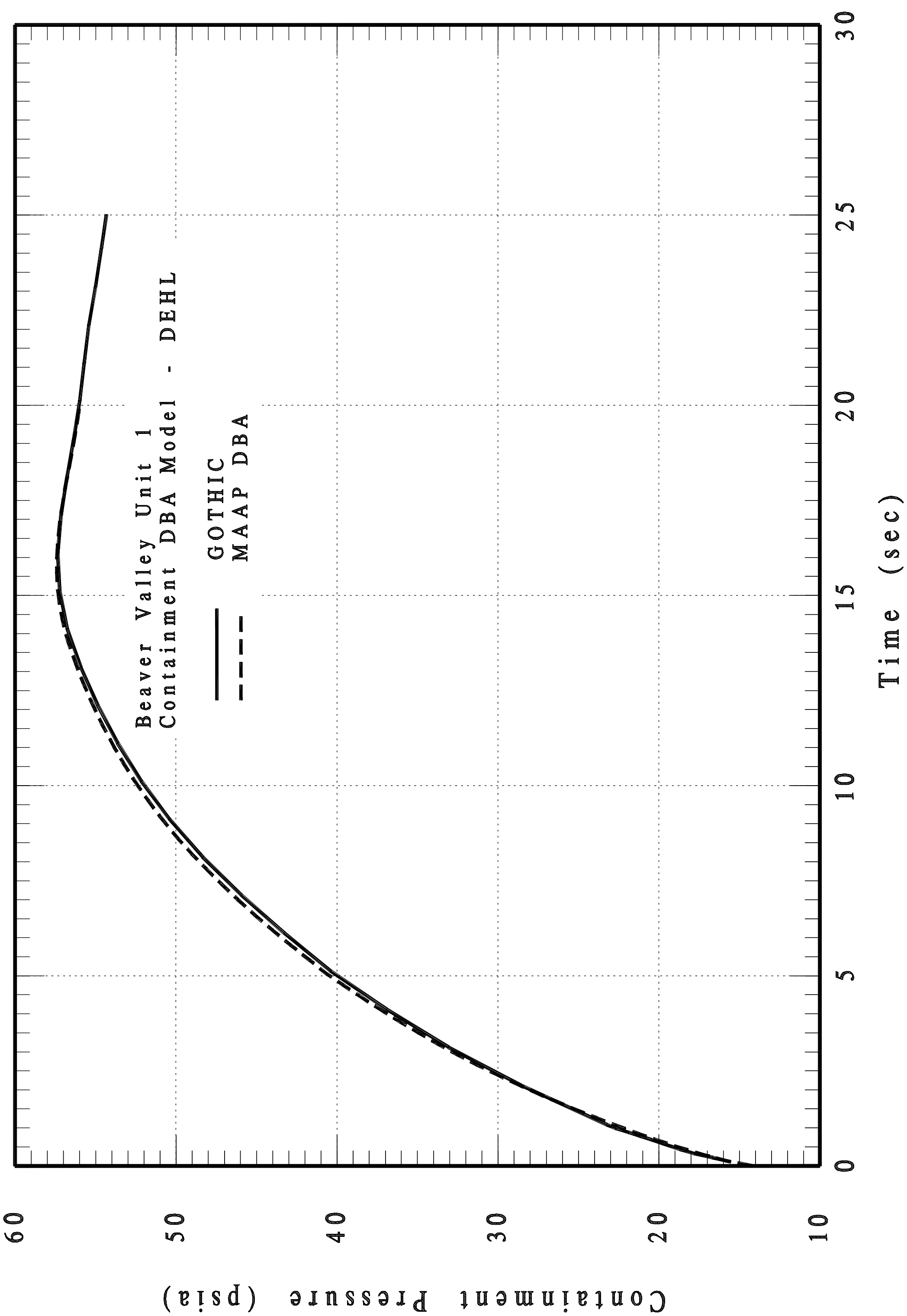


Figure 9-1
Comparison of Pressure Results from MAAP-DBA and GOTHIC for Large LOCA (BVPS Case 8L)

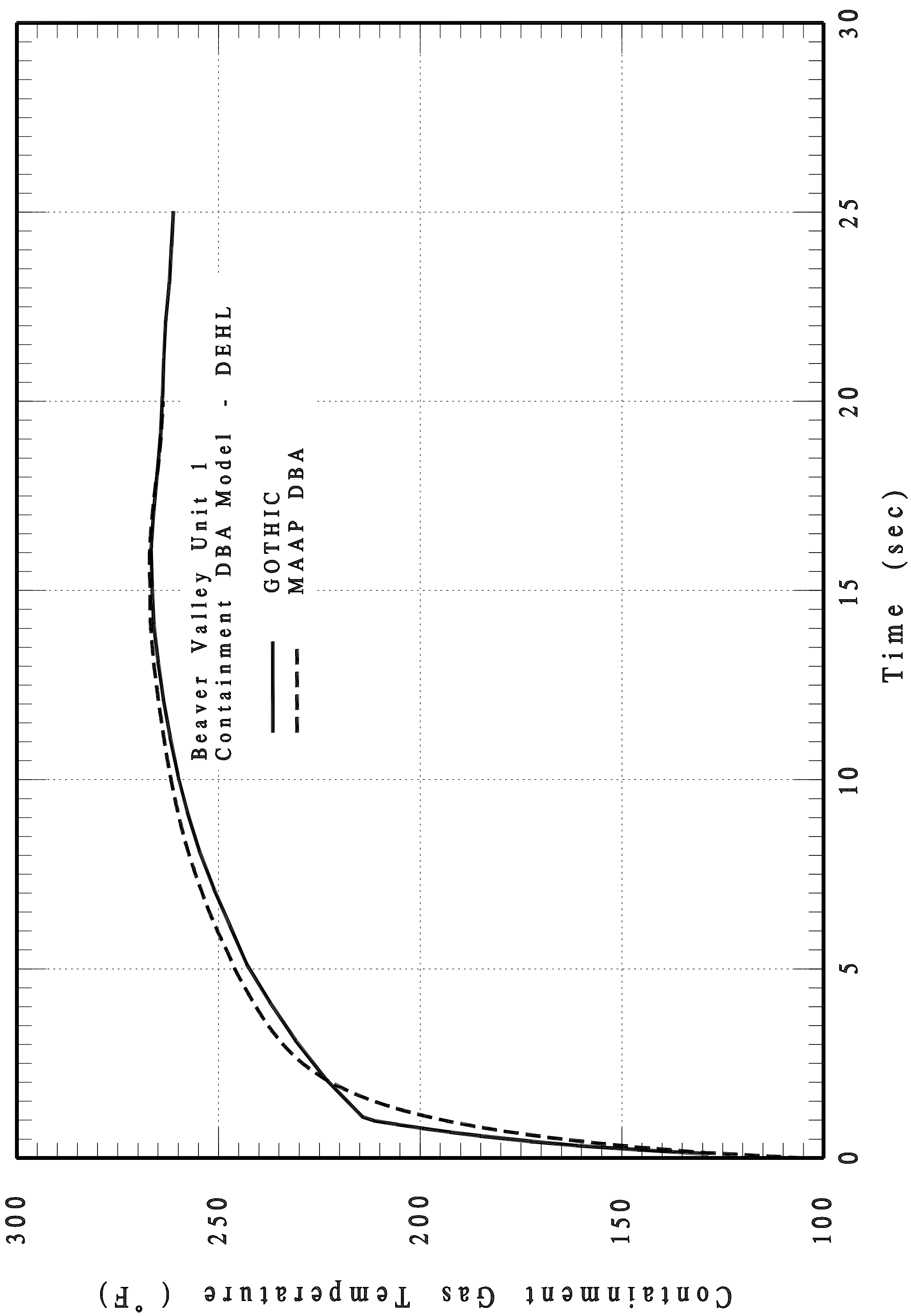


Figure 9-2
Comparison of Gas Temperature Results from MAAP-DBA and GOTHIC for Large LOCA
(BVPS Case 8L)

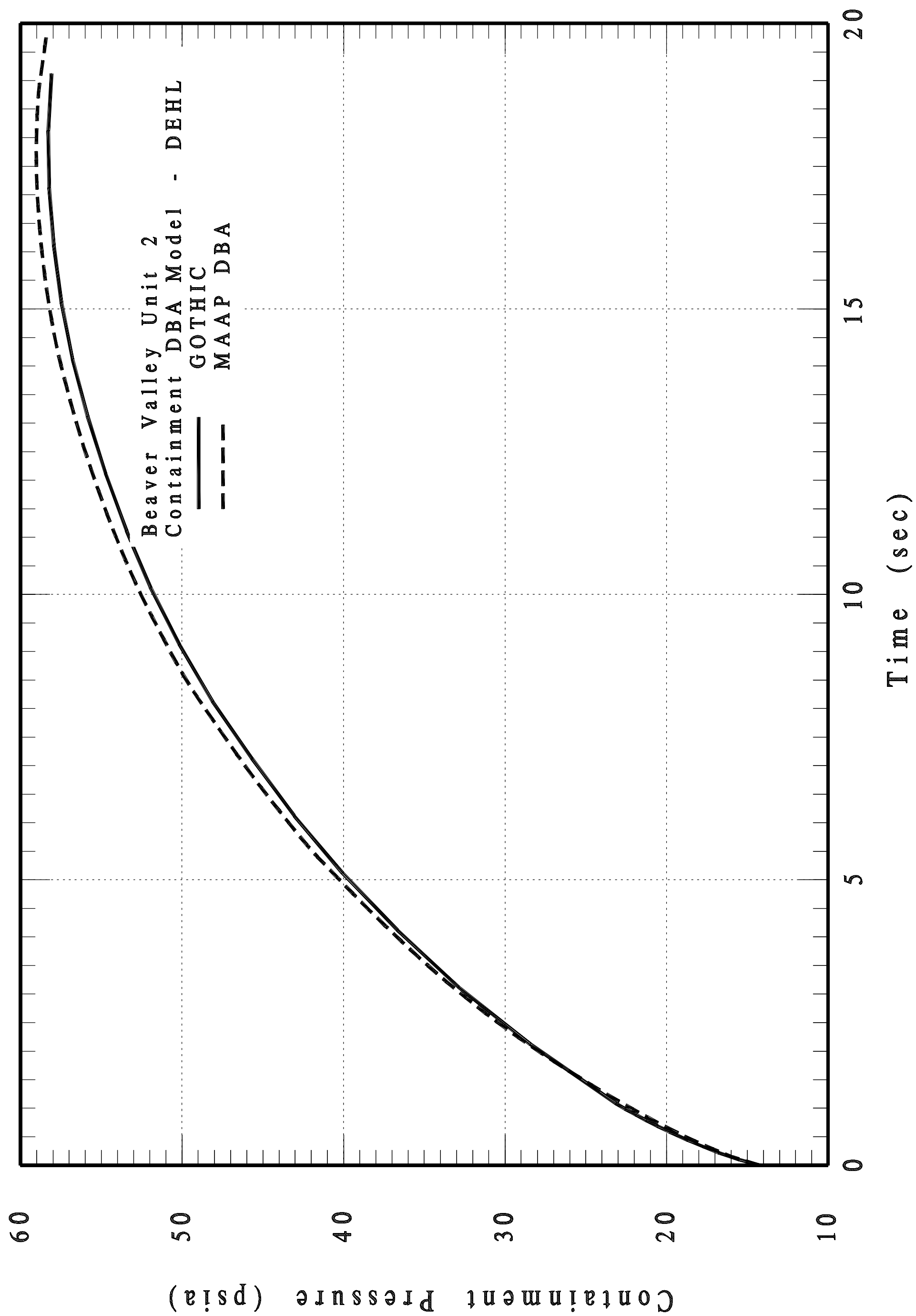


Figure 9-3
Comparison of Pressure Results from MAAP-DBA and GOTHIC for Large LOCA (BVPS Case 3L)

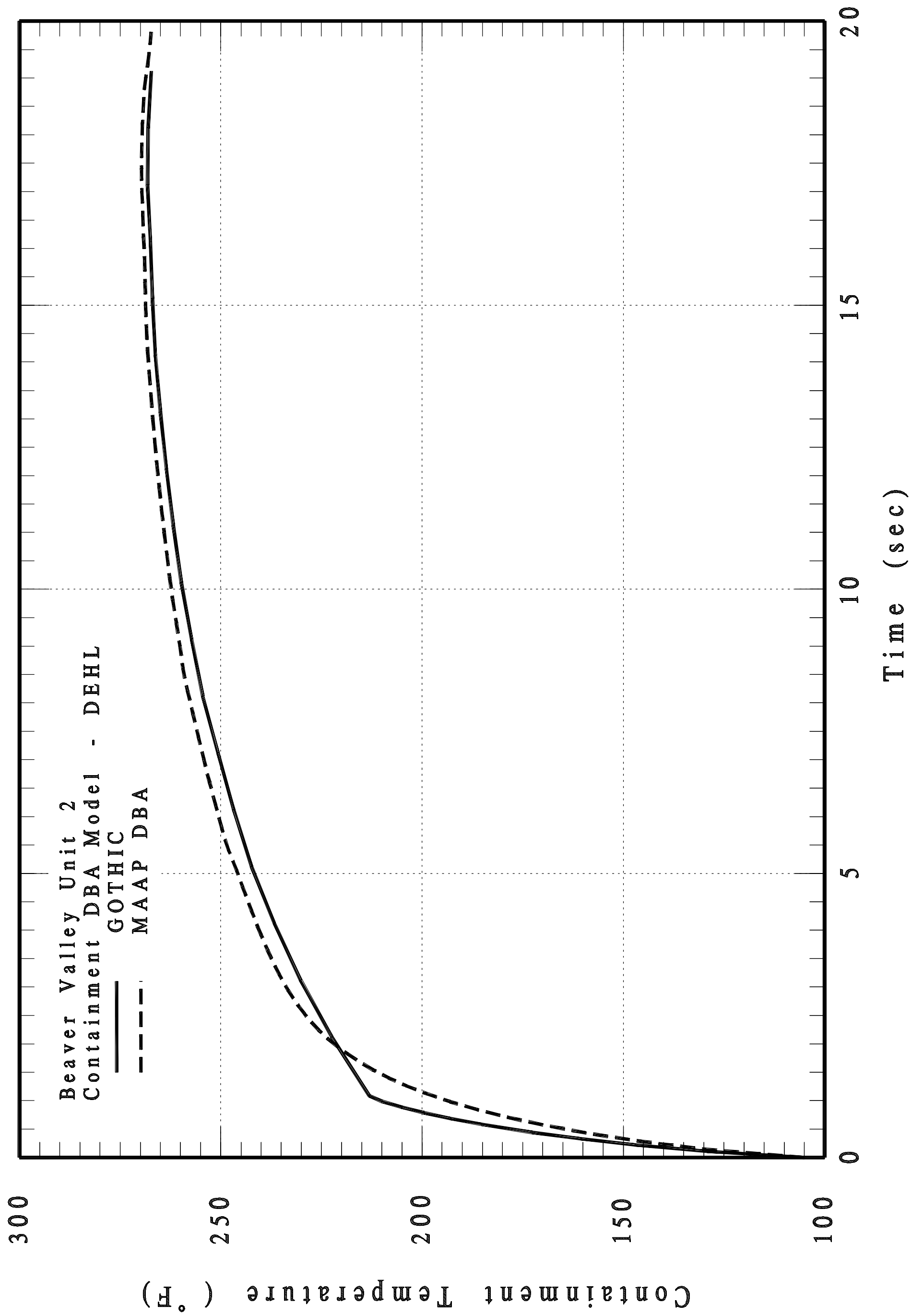


Figure 9-4
Comparison of Temperature Results from MAAP-DBA and GOTHIC for Large LOCA
(BVPS Case 3L)

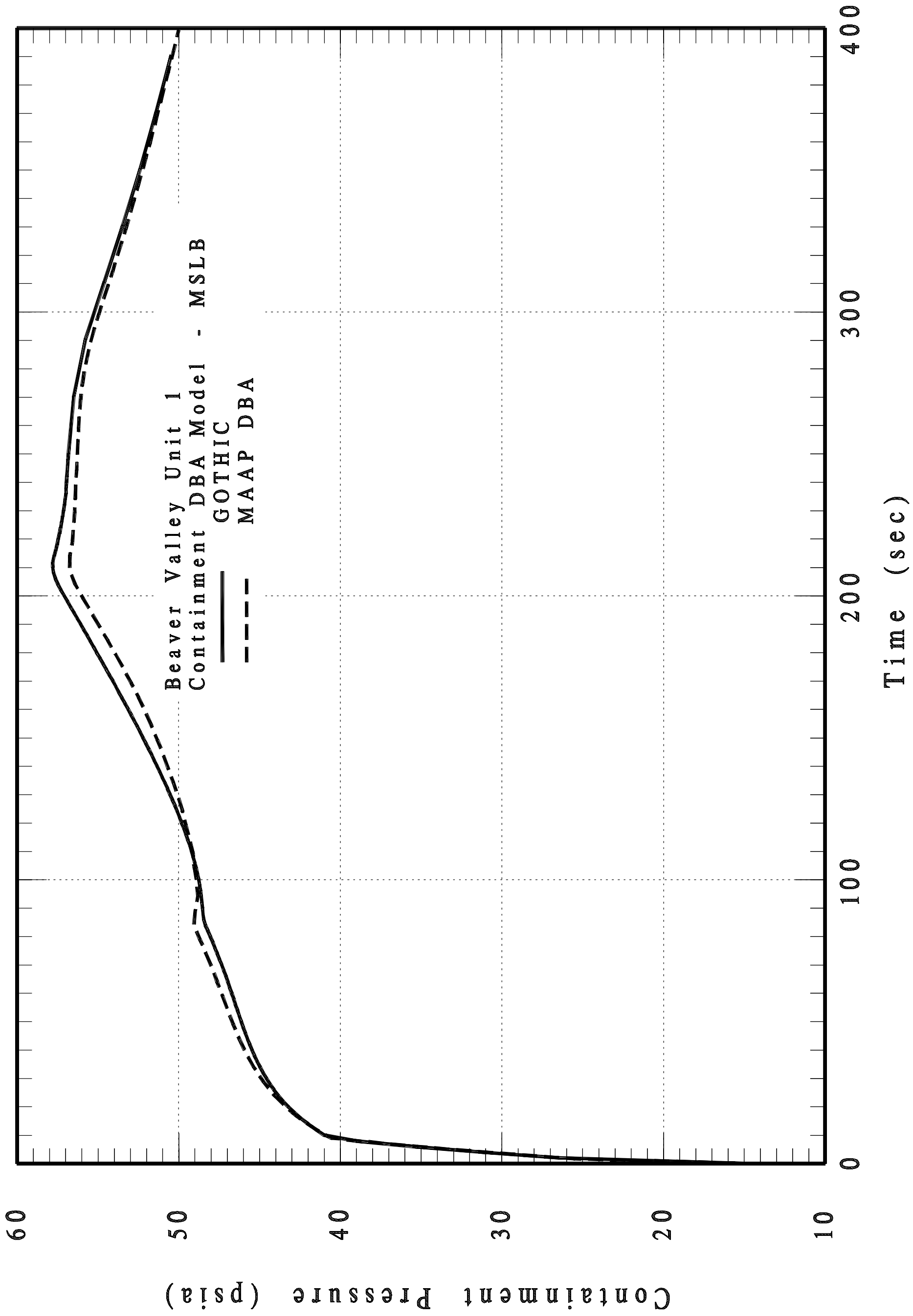


Figure 9-5
Comparison of Pressure Results from MAAP-DBA and GOTHIC for MSLB (BVPS Case 15M)

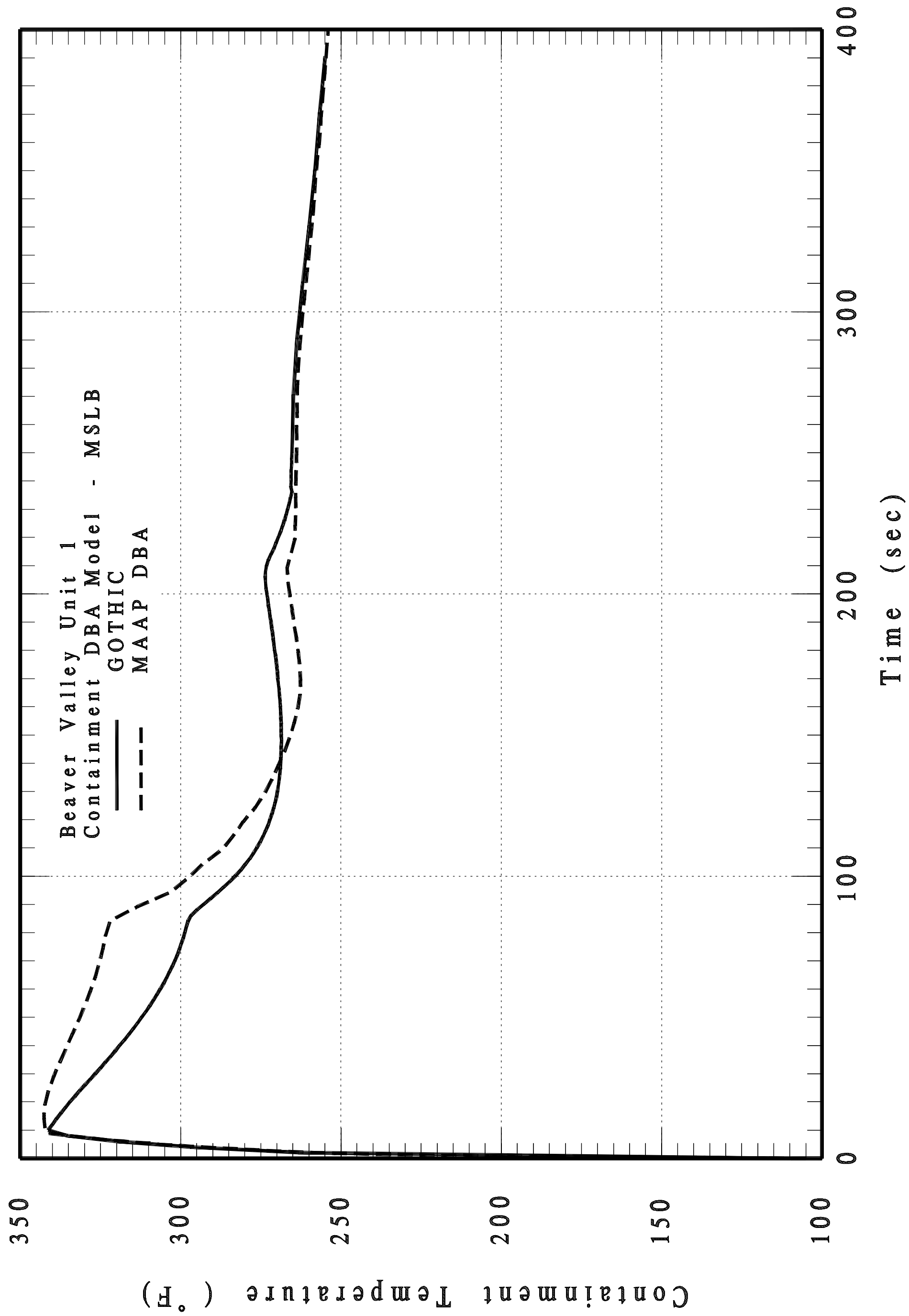


Figure 9-6
Comparison of Temperature Results from MAAP-DBA and GOHTIC for MSLB (BVPS Case 15M)

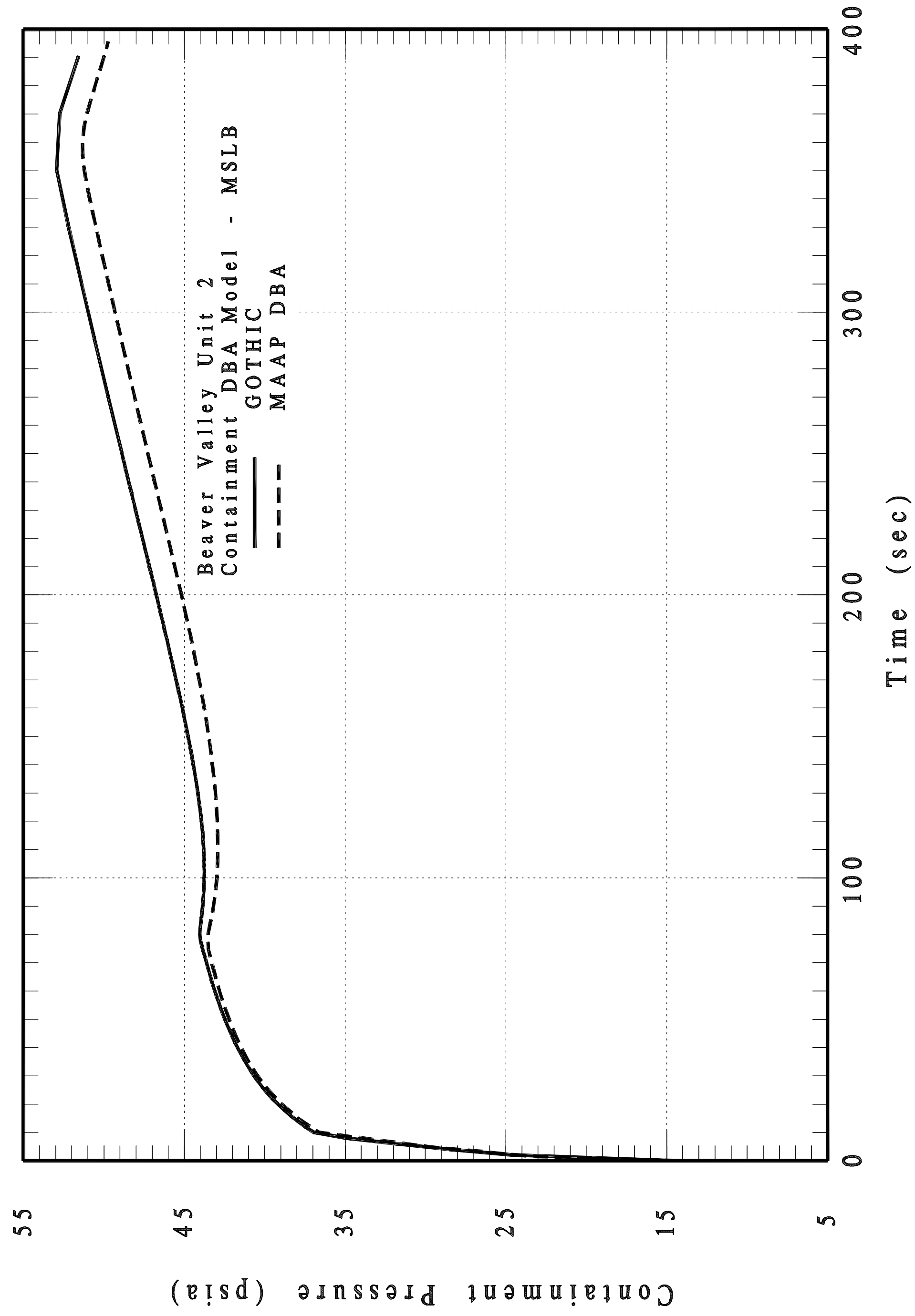


Figure 9-7
Comparison of Pressure Results from MAAP-DBA and GOTHIC for MSLB (BVPS Case 16M)

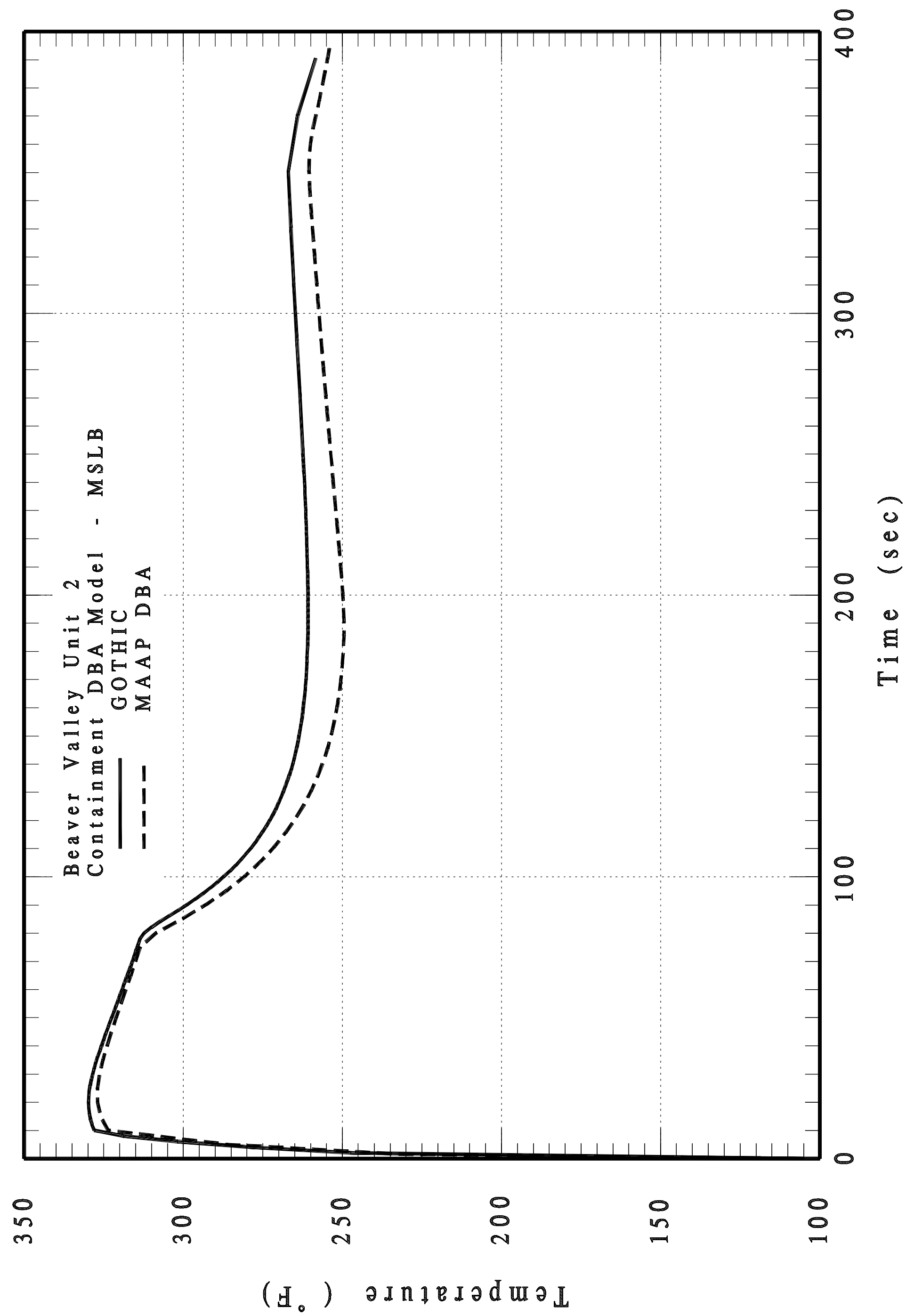


Figure 9-8
Comparison of Temperature Results from MAAP-DBA and GOTHIC for MSLB (BVPS Case 16M)

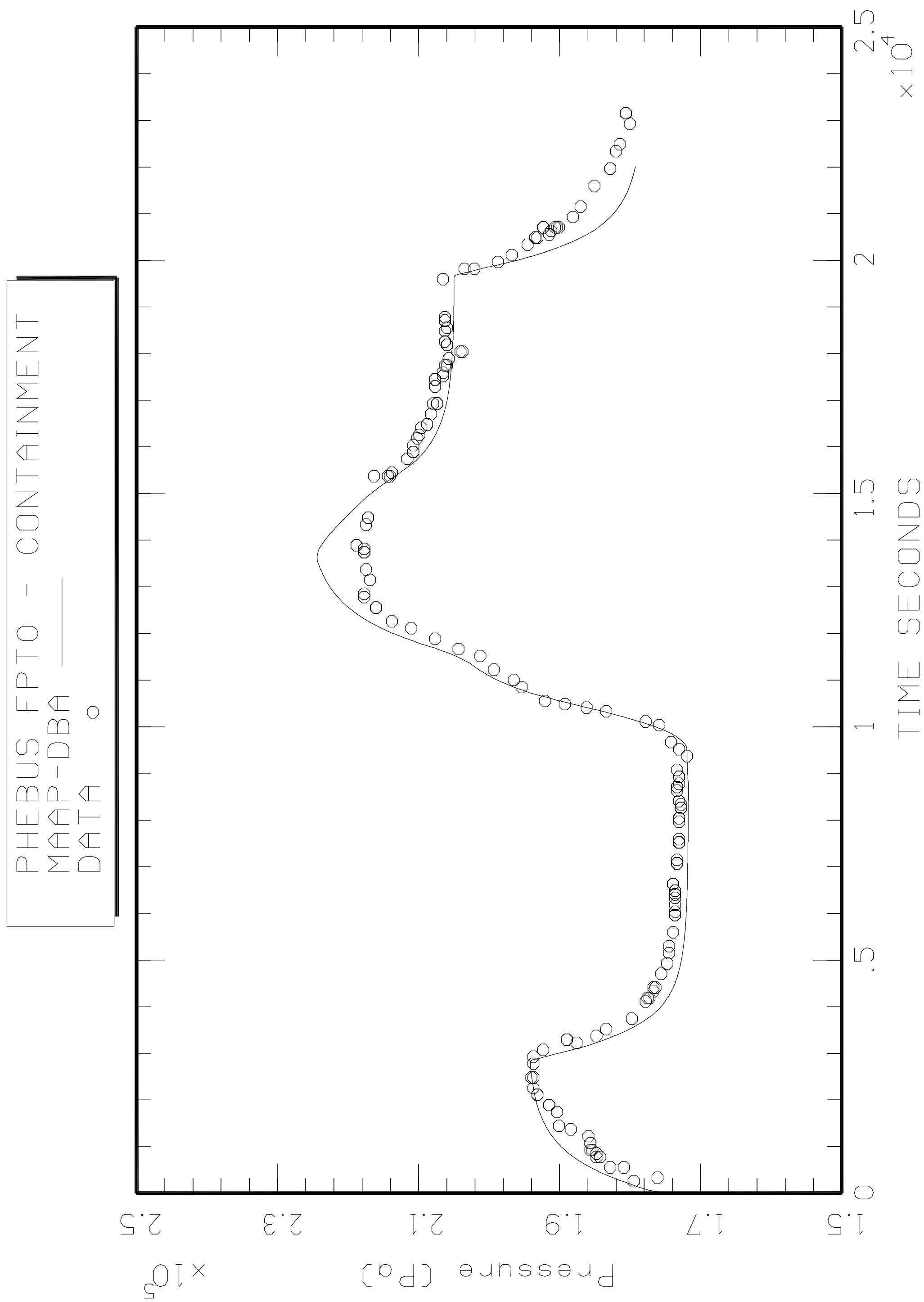


Figure 9-9
PHEBUS FTP0 Pressure Profile

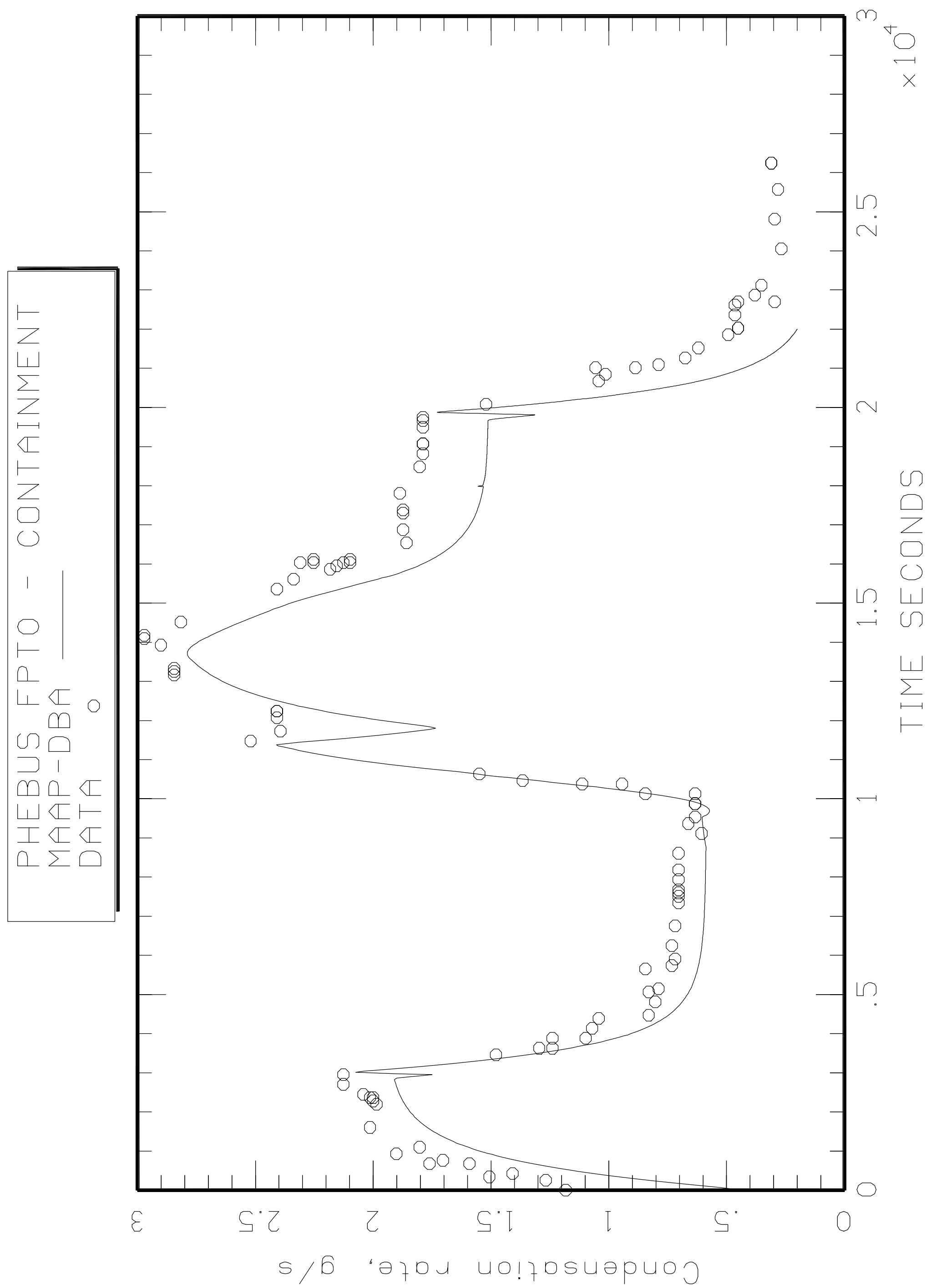


Figure 9-10
PHEBUS FPT0 Condensation Rate Profile

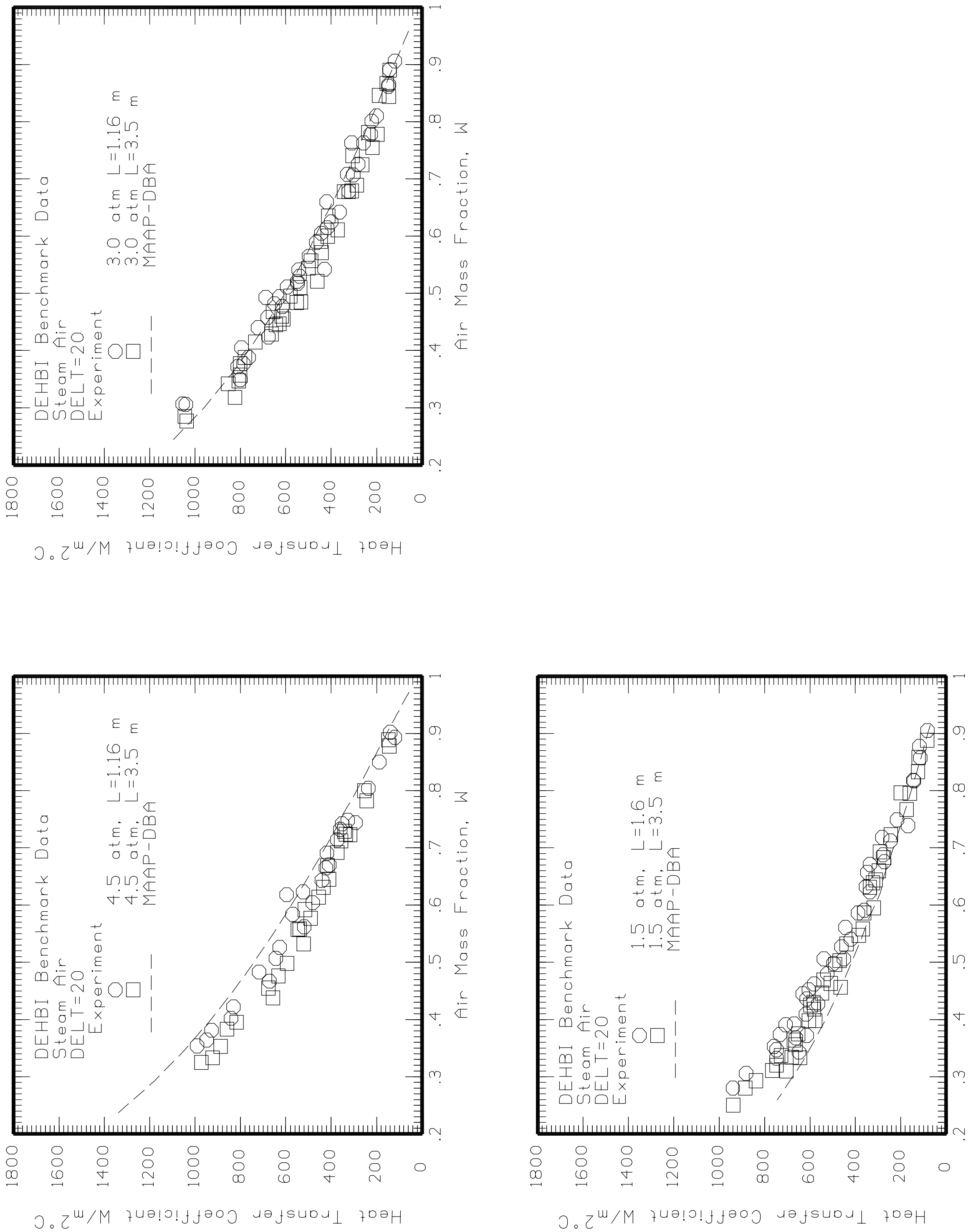


Figure 9-11
Comparison of the MAAP-DBA Condensation Heat Transfer Model with the Experimentally Determined Steam-Air Condensing Heat Transfer Coefficients

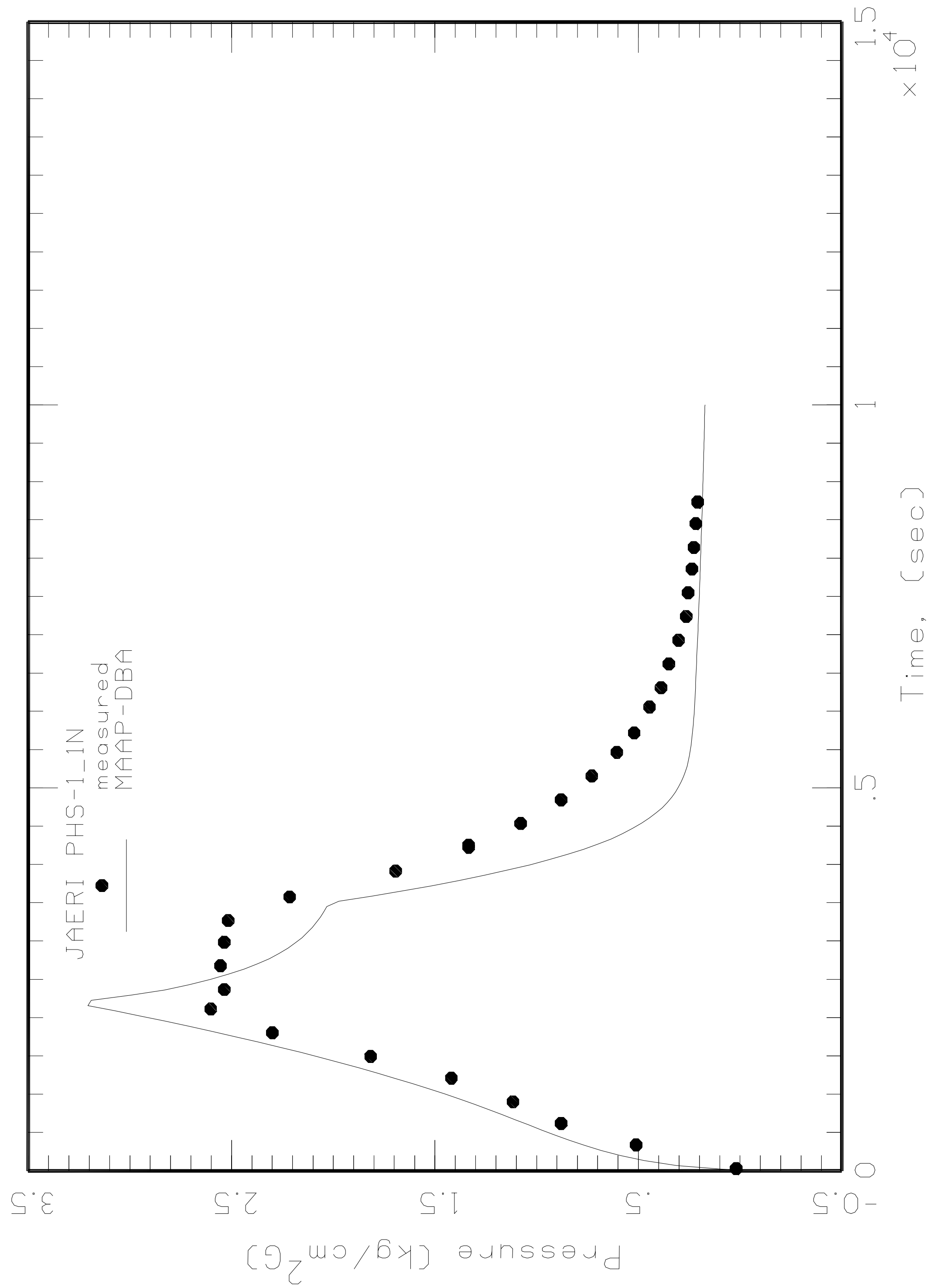


Figure 9-12
MAAP-DBA Single Node Model Pressure Profile for JAERI Test PHS-1

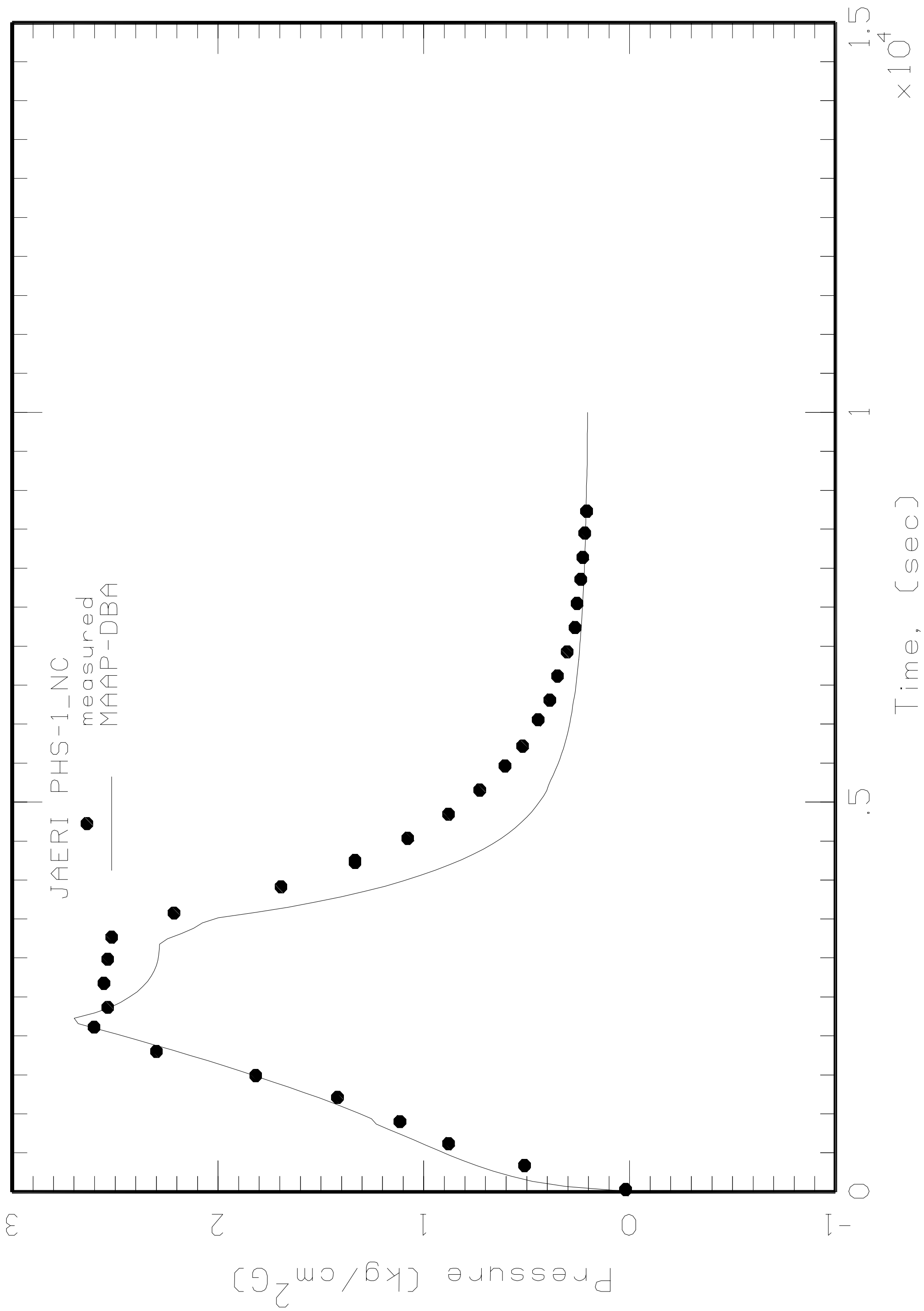


Figure 9-13
MAAP-DBA Multiple Node Model Pressure Profile for JAERI Test PHS-1

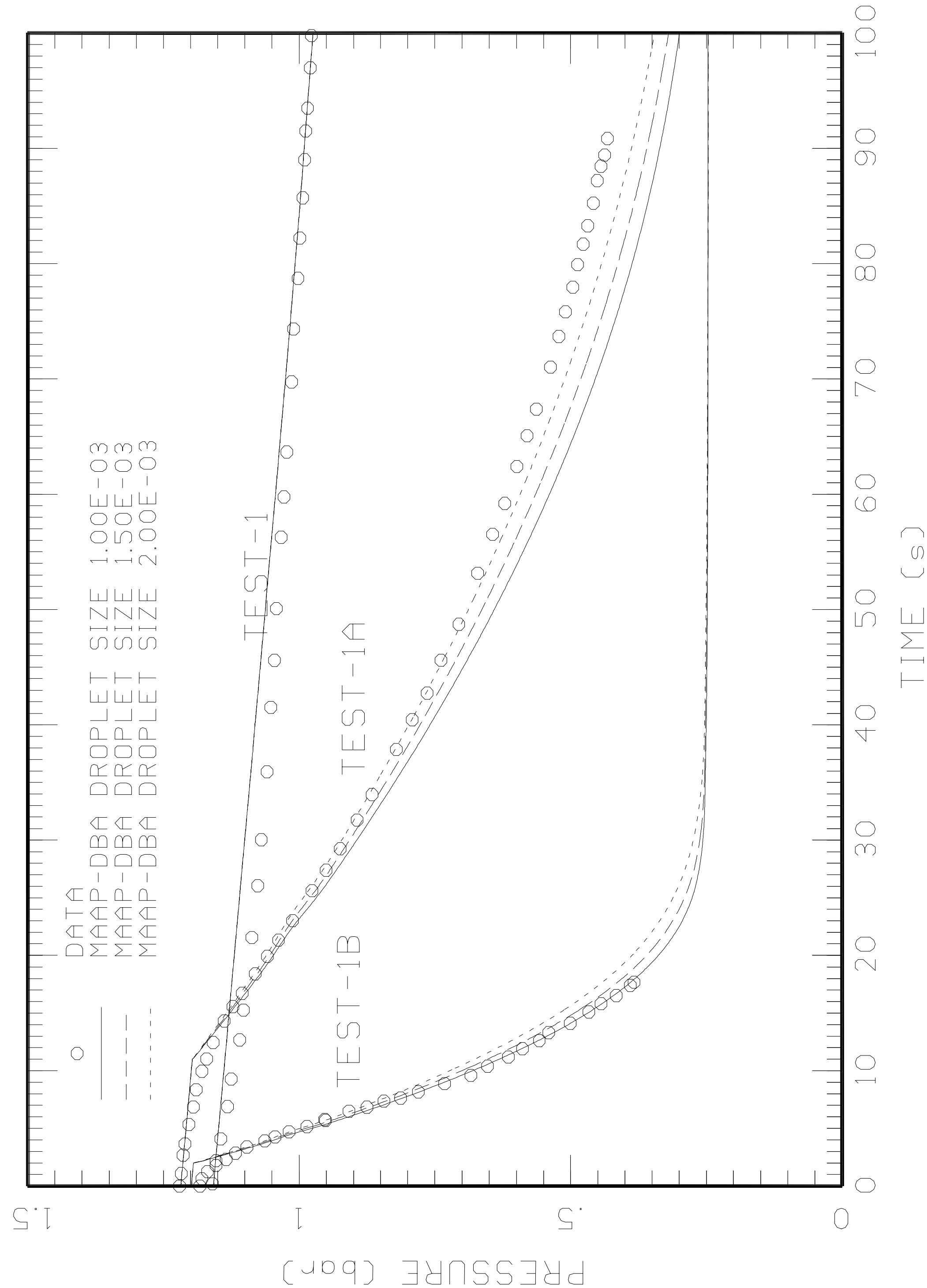


Figure 9-14
MAAP-DBA Pressure Profiles for Kulic Spray Tests

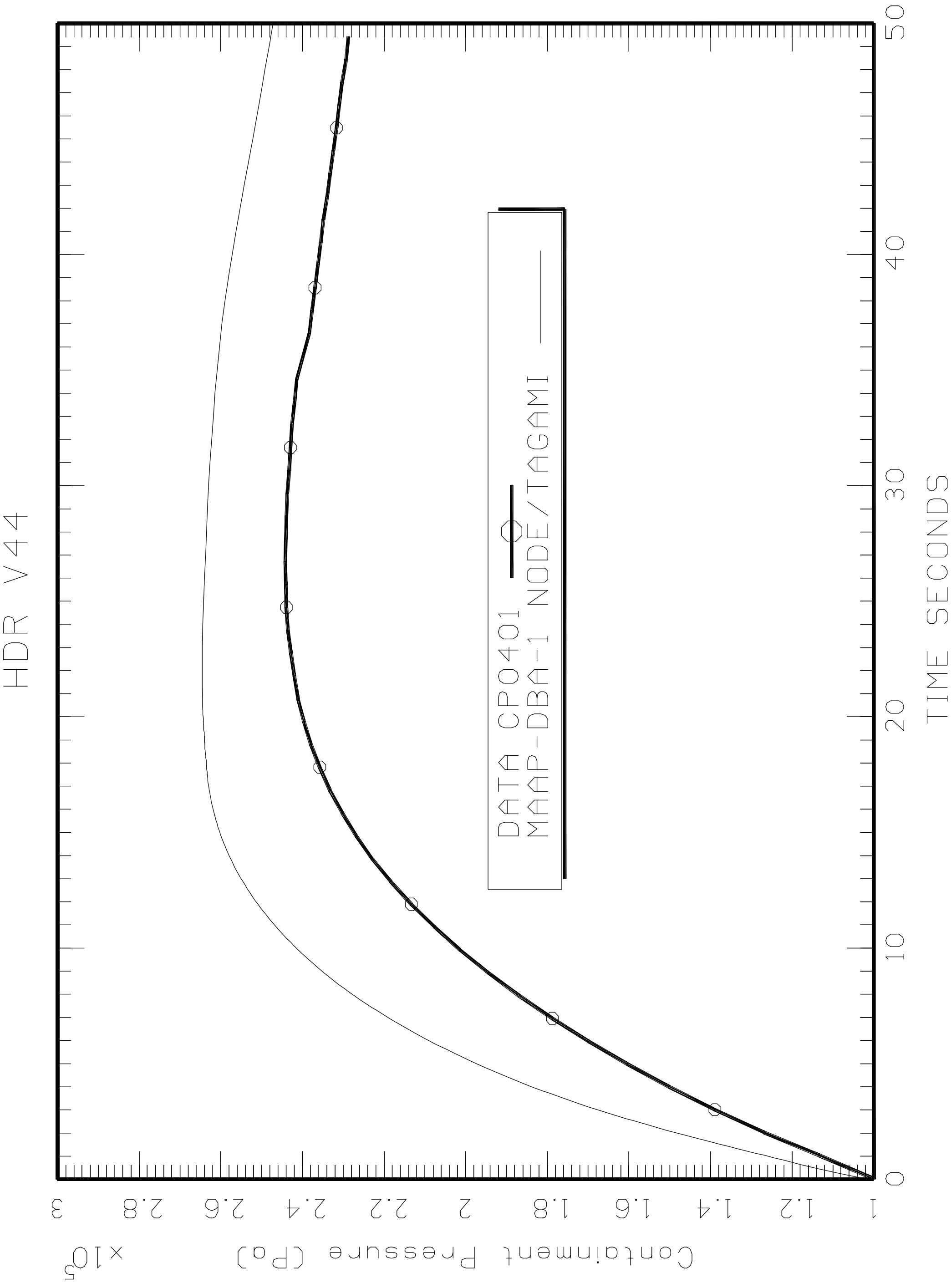


Figure 9-15
HDR-V44 Pressure Profile

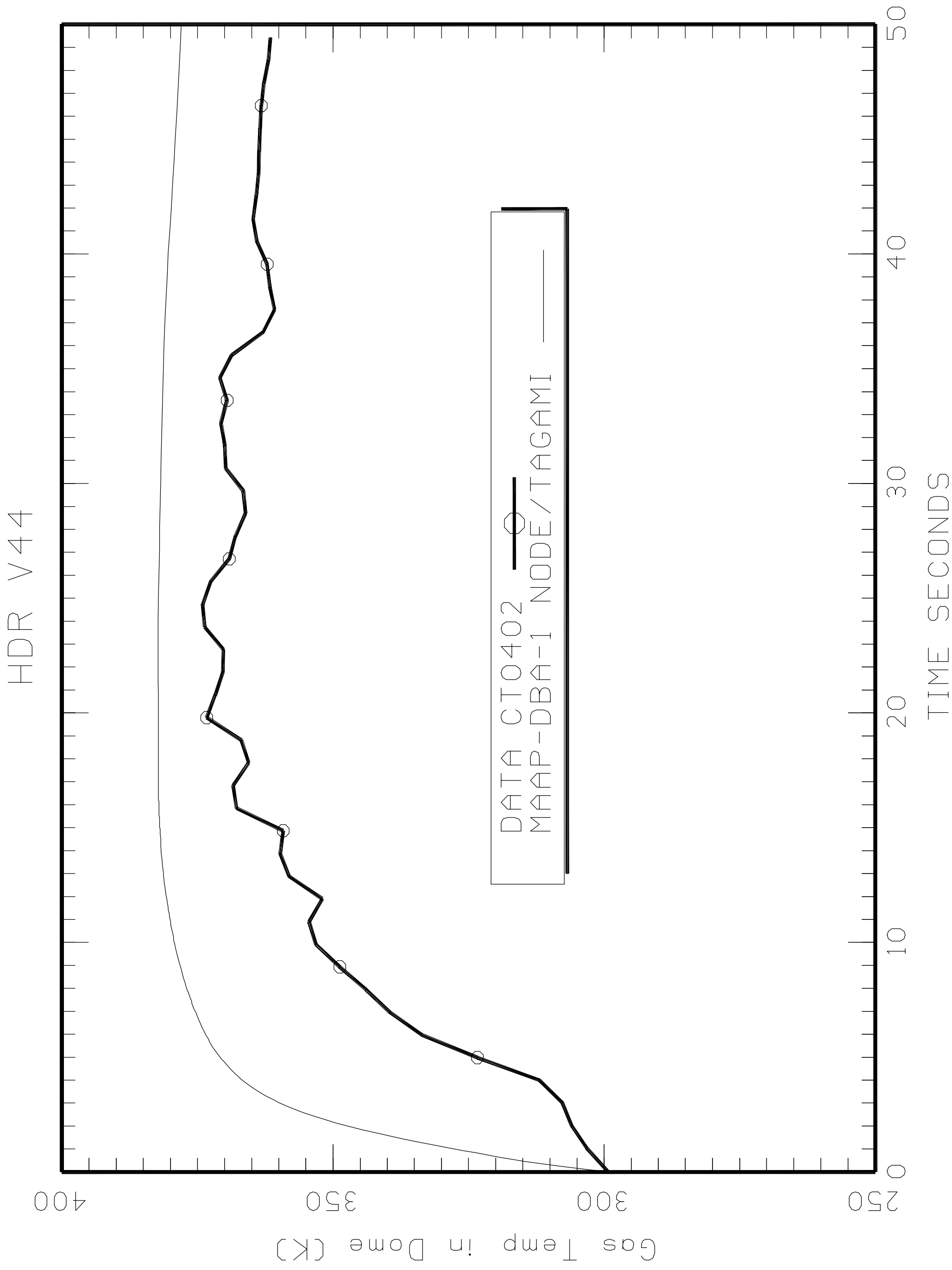


Figure 9-16
HDR-V44 Gas Temperature Profile

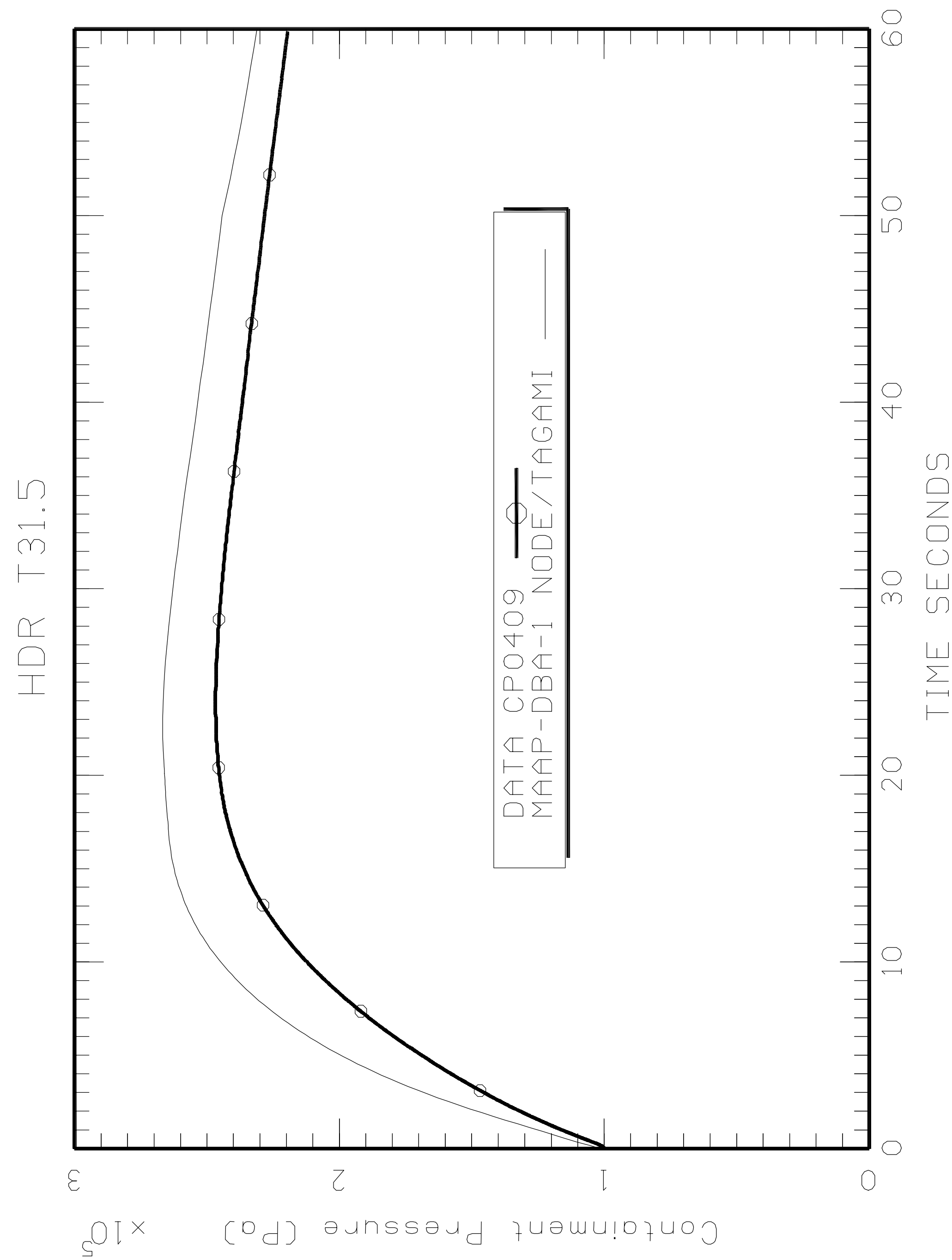


Figure 9-17
HDR-T31.5 Pressure Profile

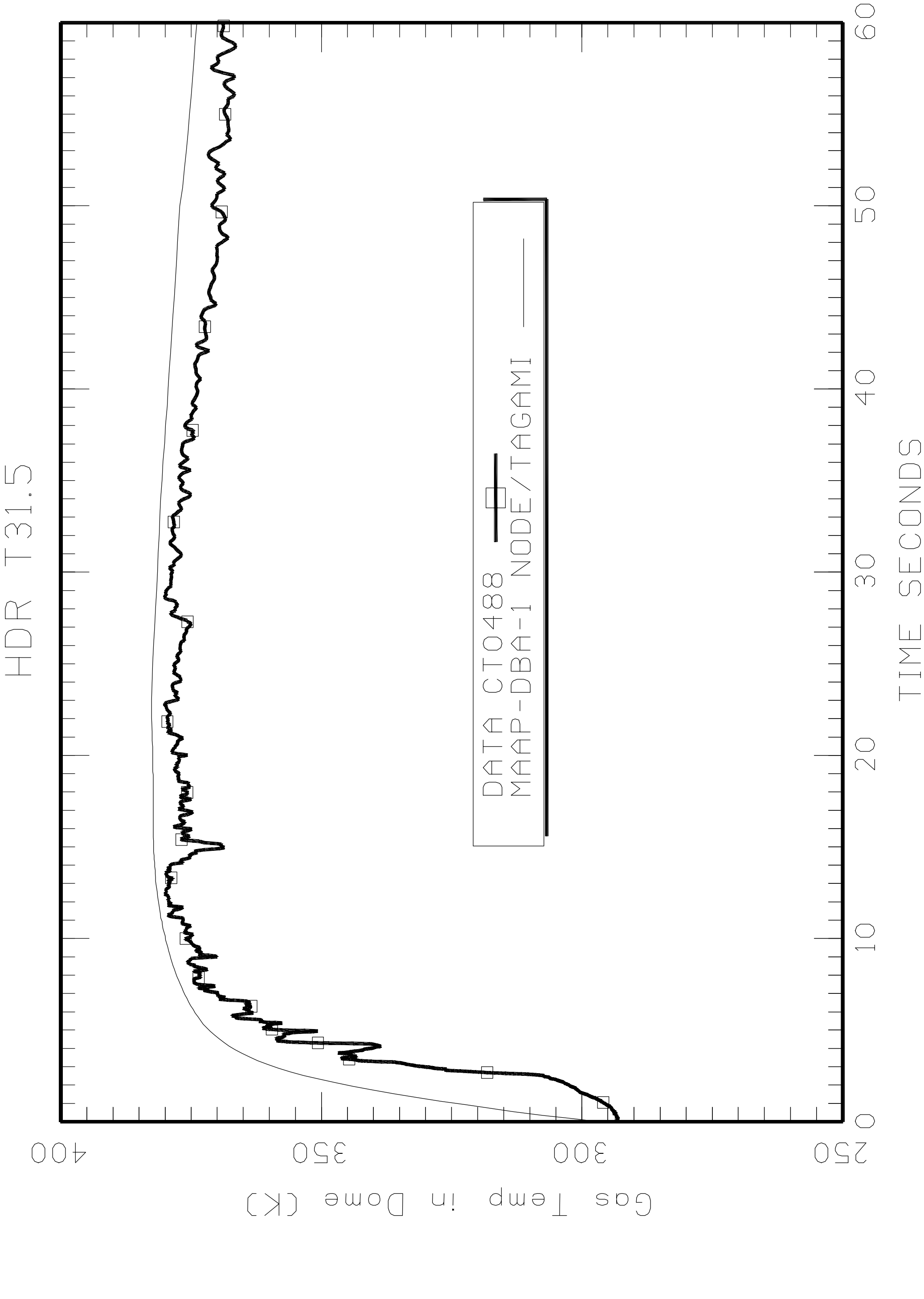
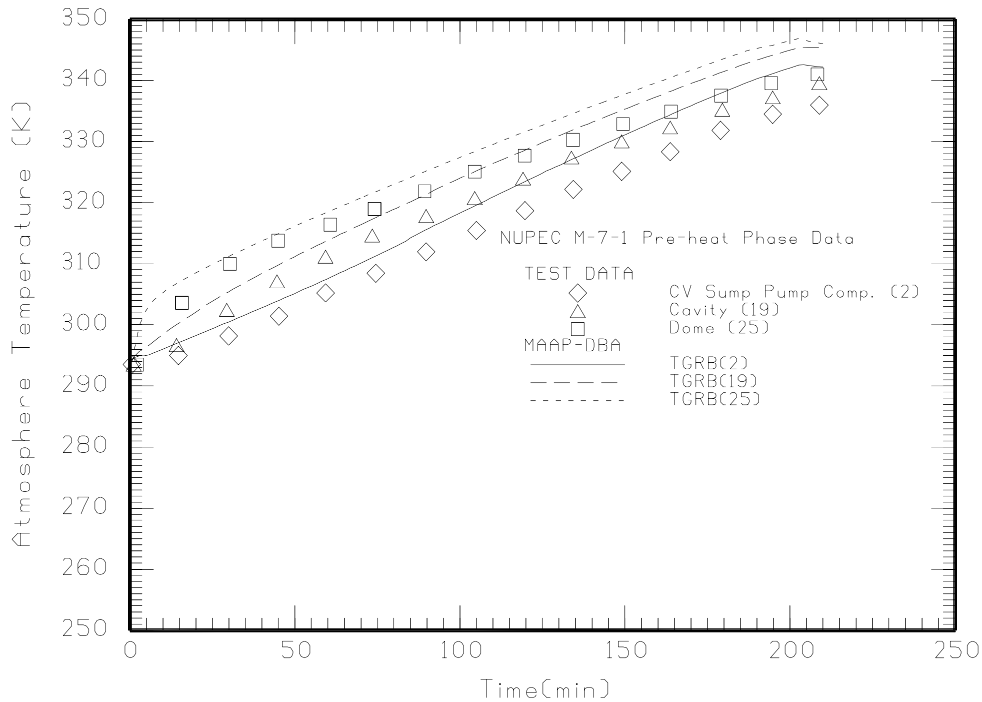


Figure 9-18
HDR-T31.5 Gas Temperature Profile



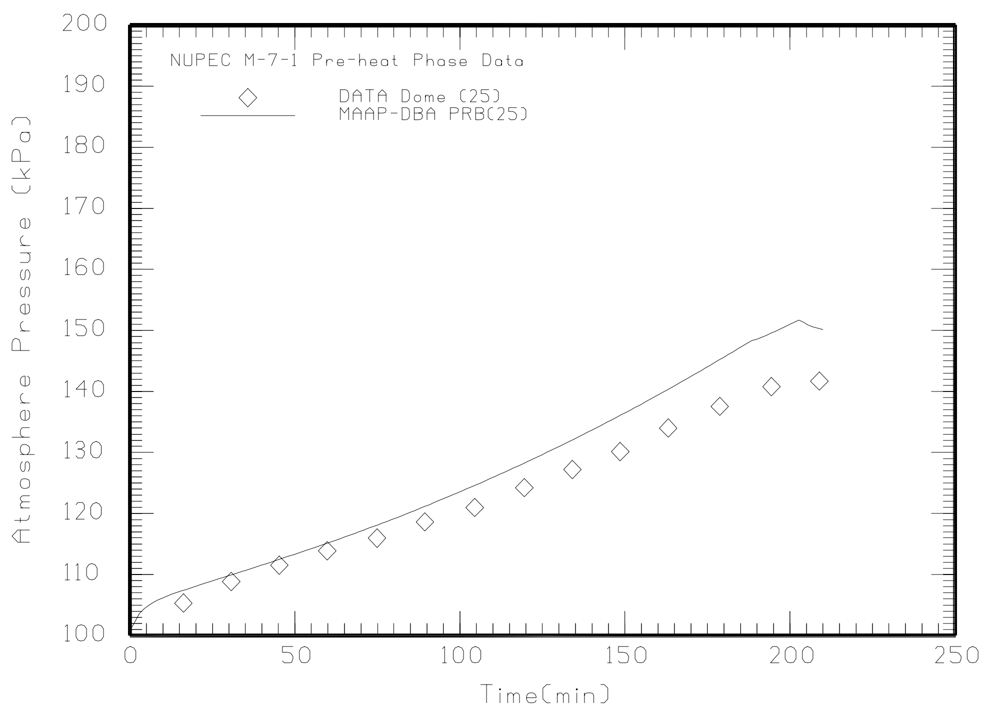


Figure 9-19
Comparison of NUPEC M-7-1 Preheat Phase Gas Temperatures and Containment Pressure

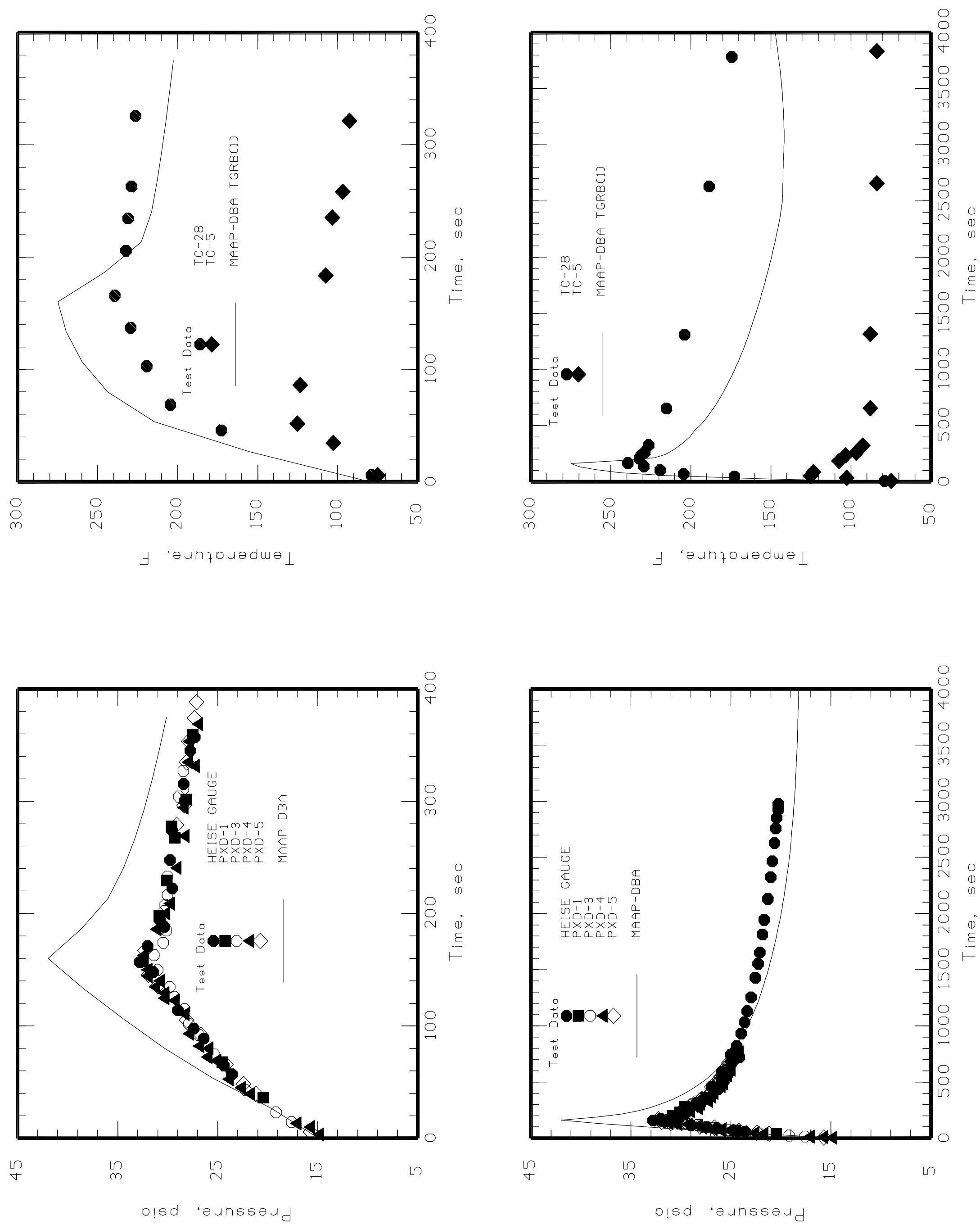


Figure 9-20
Comparison of CVTR Test 3 Containment Pressure

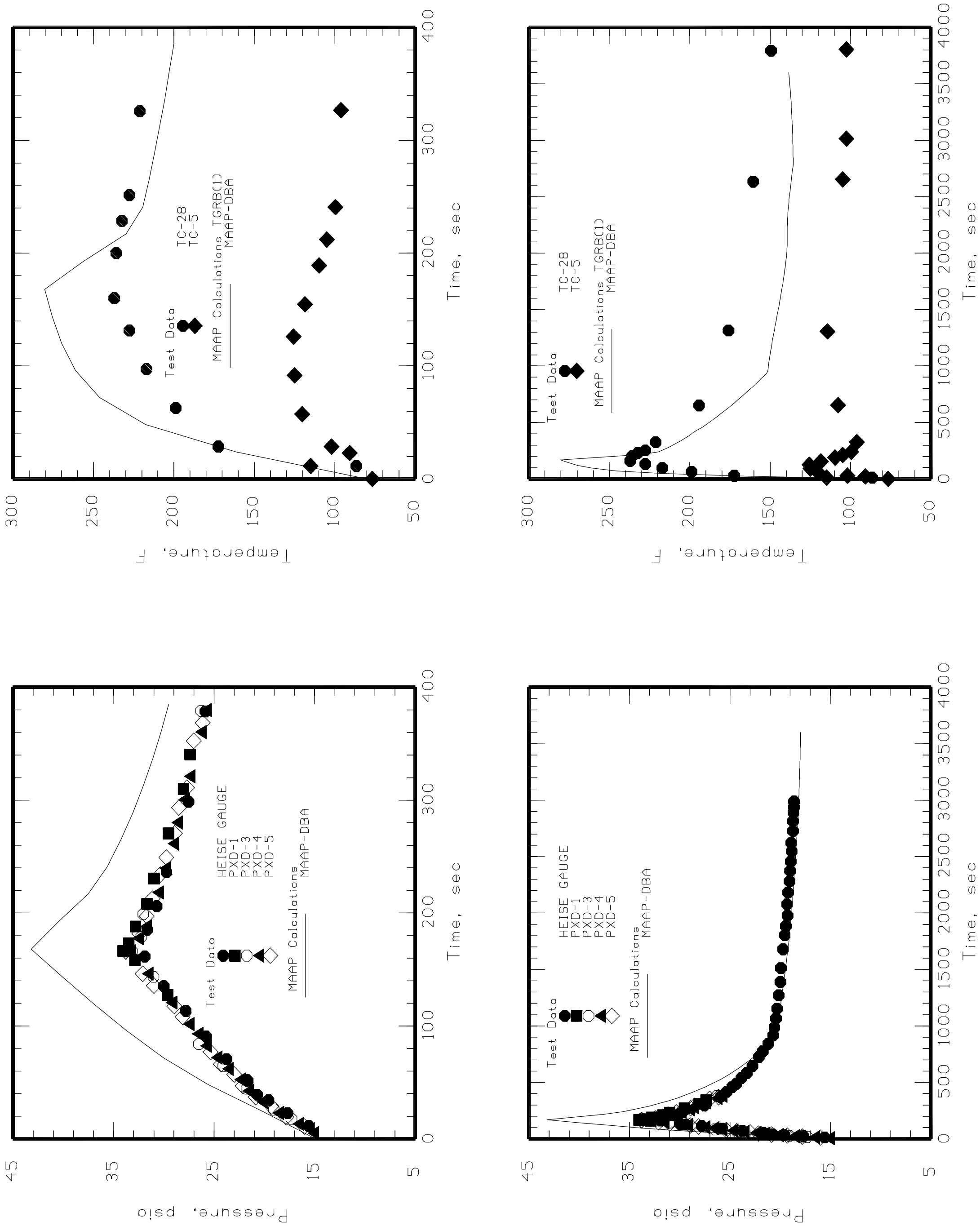


Figure 9-21
Comparison of CVTR Test 4 Containment Pressure

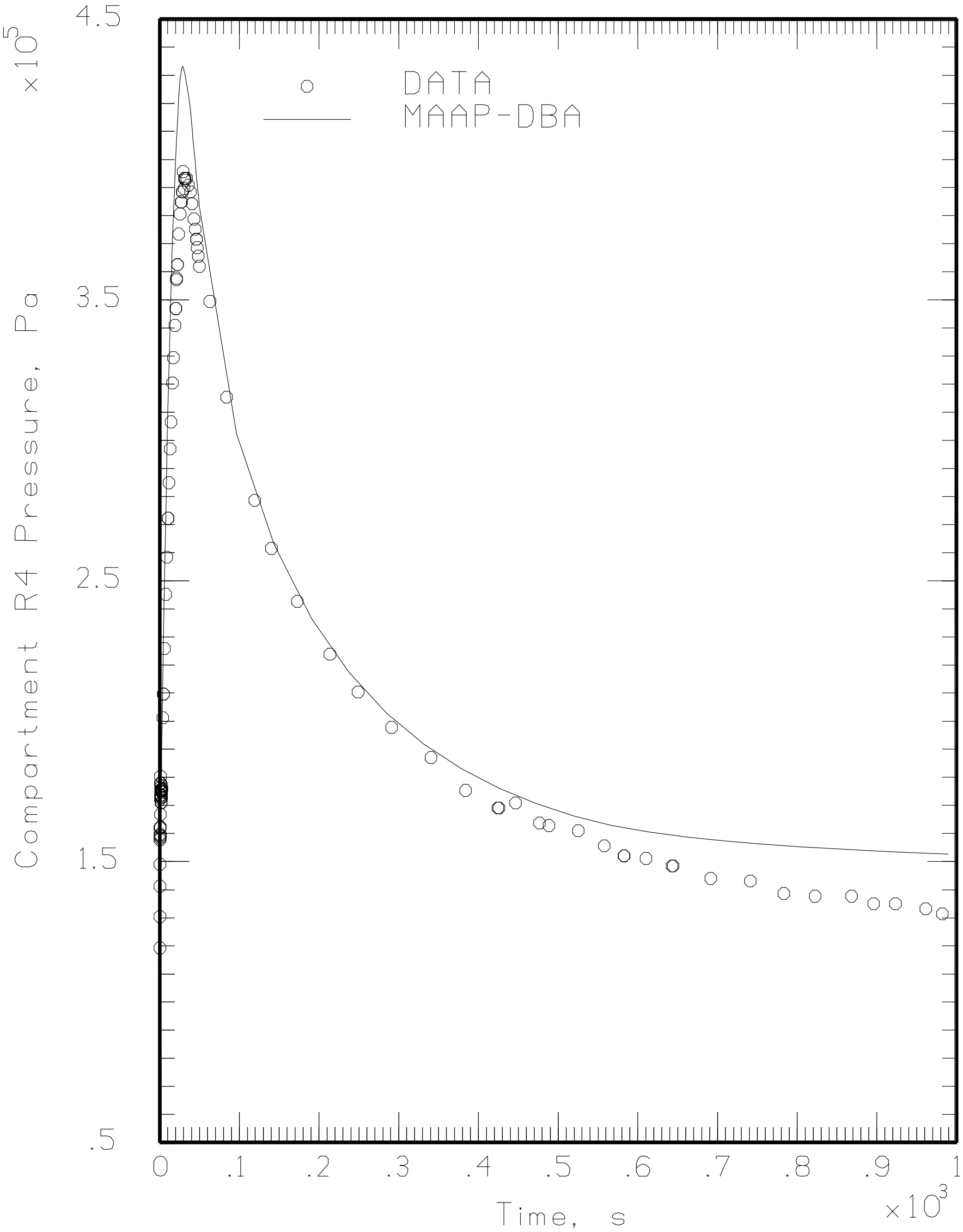


Figure 9-22
Comparison of BFMC D-16 Pressure History in the Break Compartment

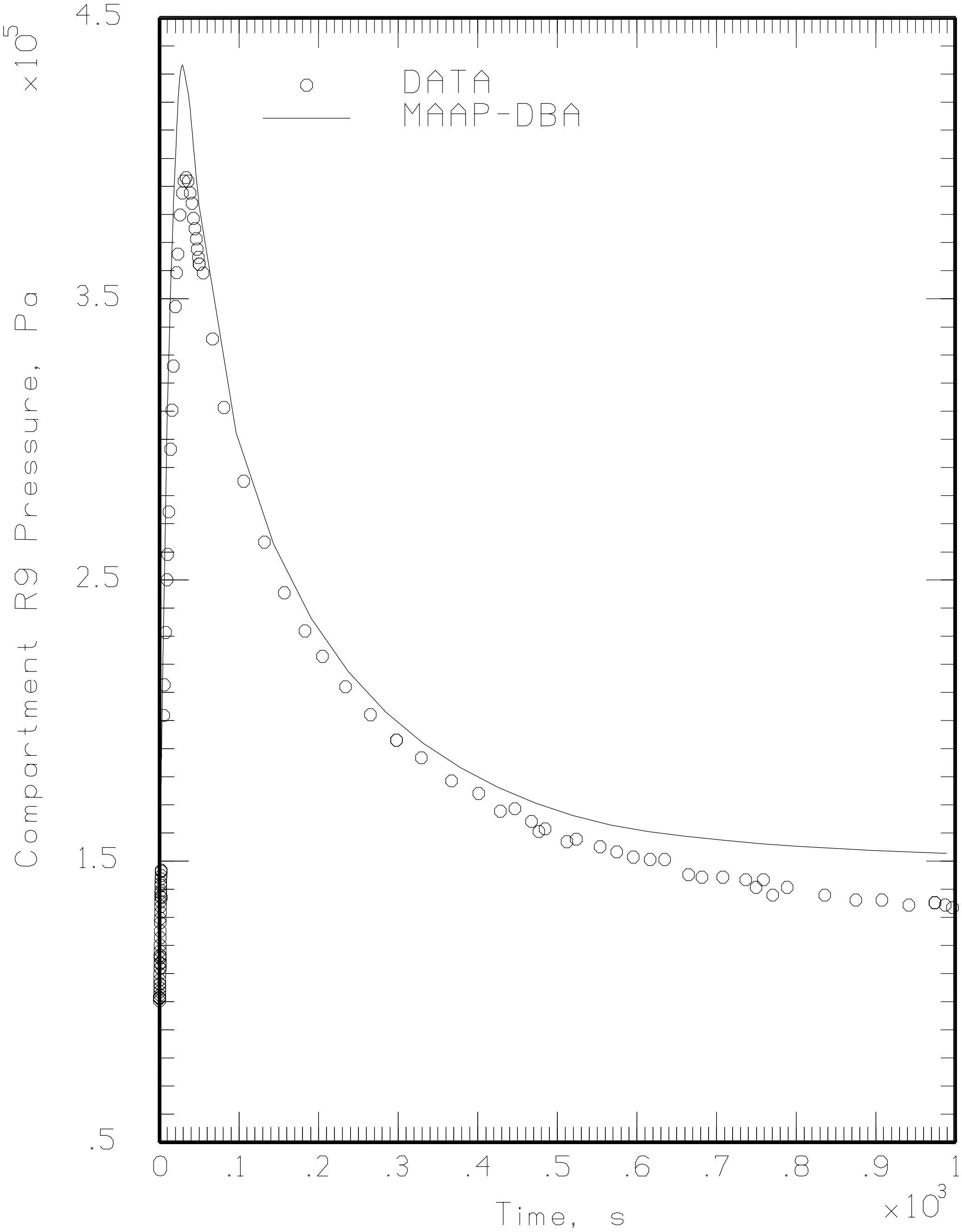


Figure 9-23
Comparison of BFMC D-16 Pressure History for Outer Room

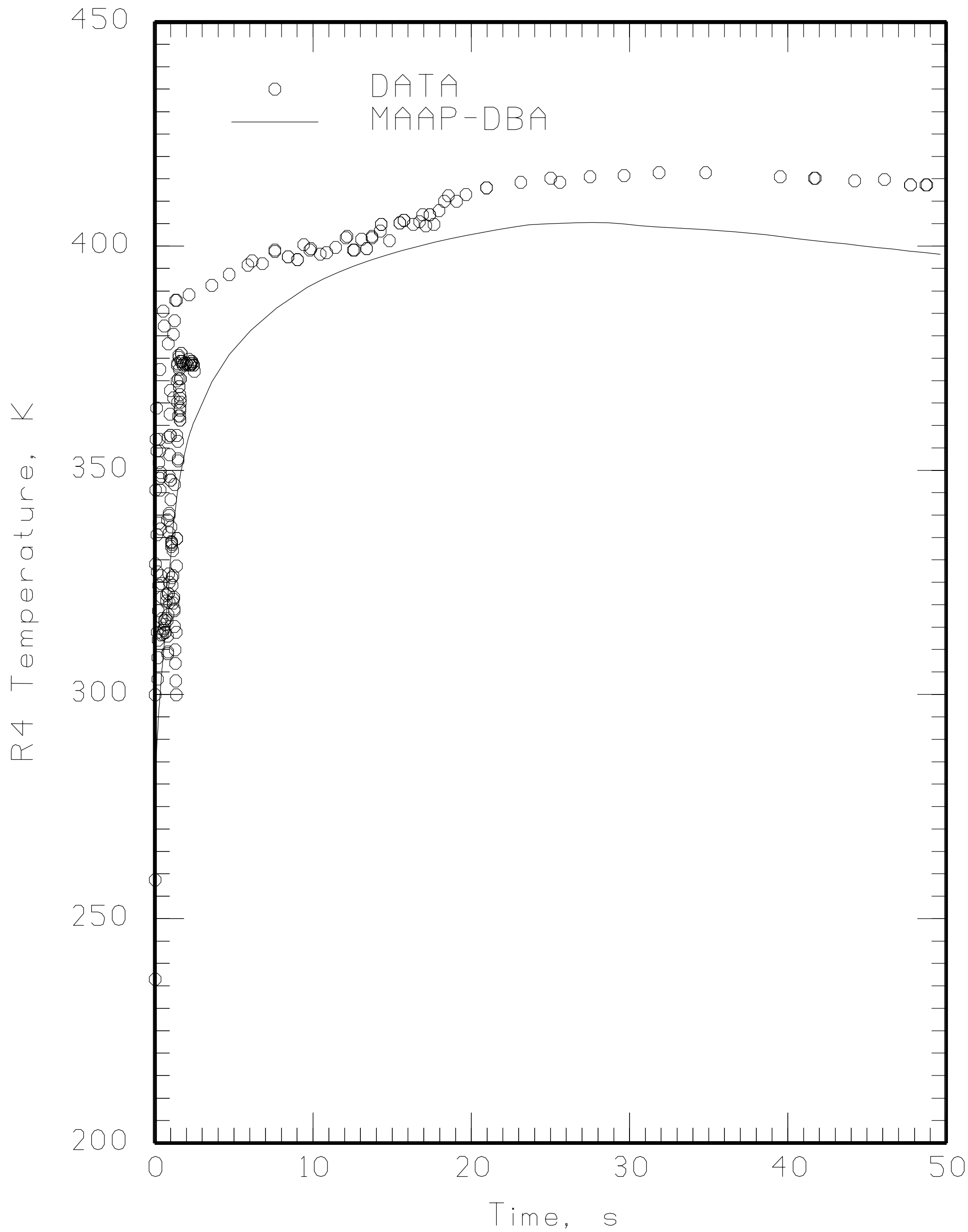


Figure 9-24
Comparison of BFMC D-16 Temperature History for Break Room

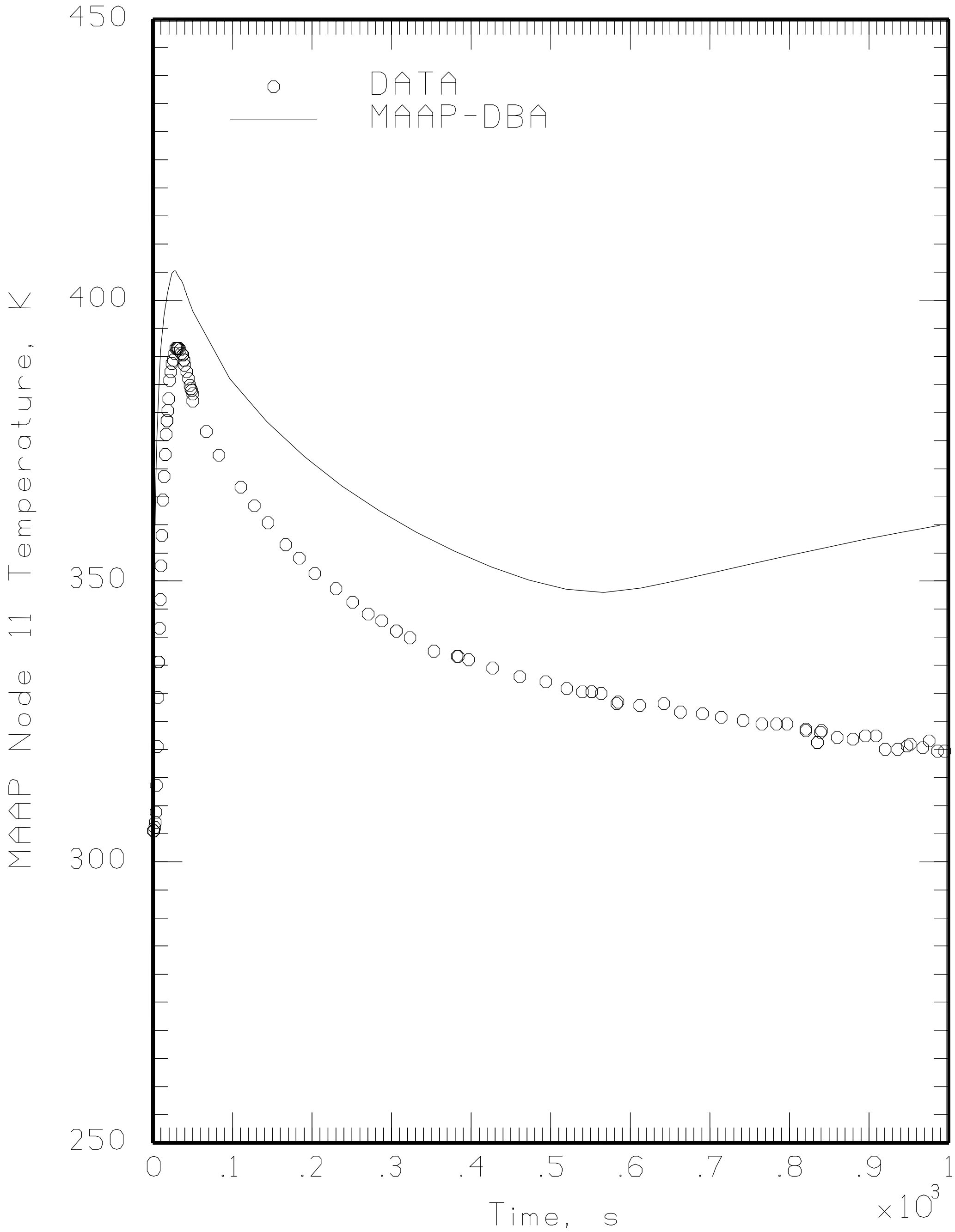


Figure 9-25
Comparison of BFMC D-16 Temperature History for Outer Room

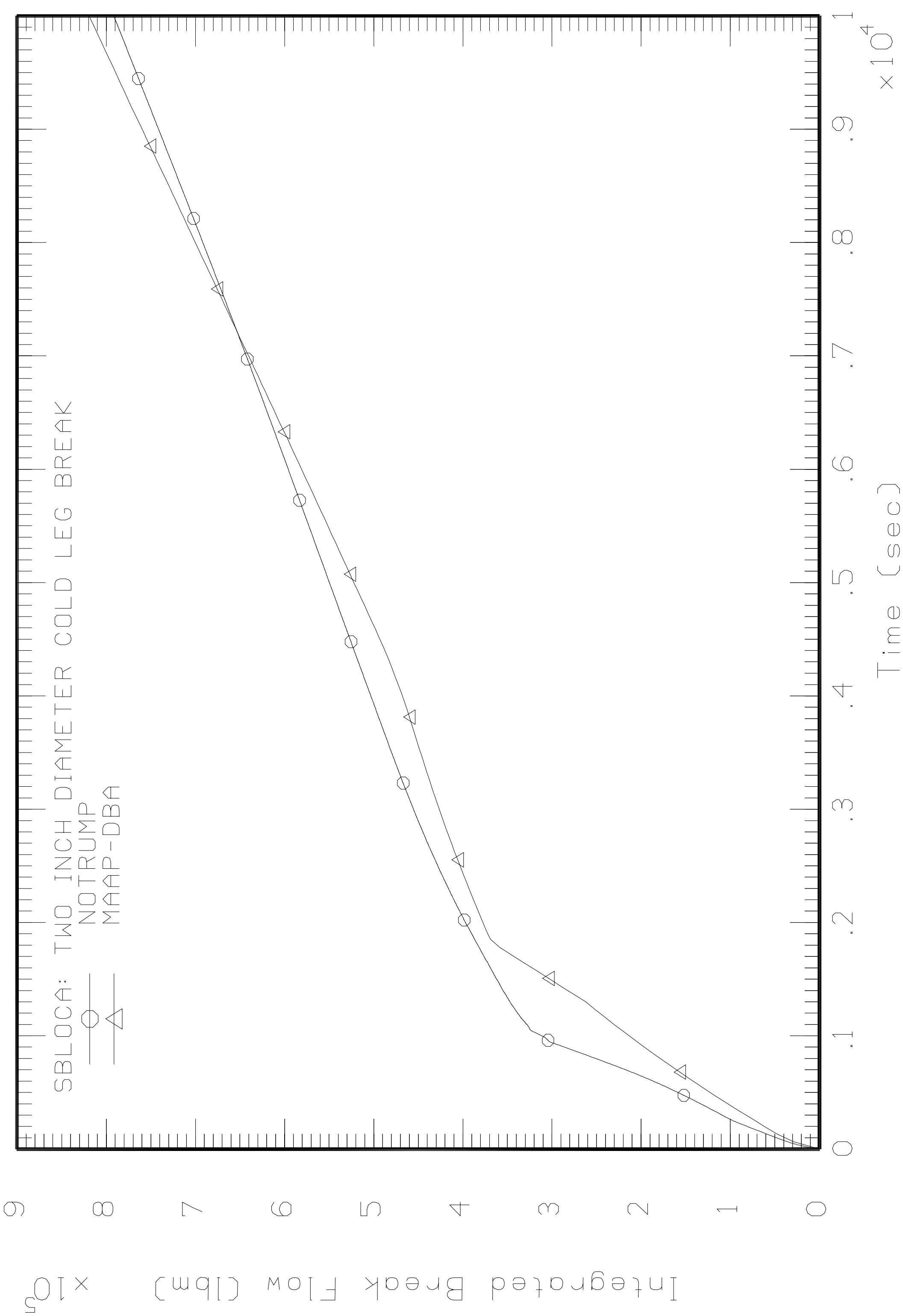


Figure 9-26a
Comparison of 2 Inch Cold Leg Releases for SBLOCA Analysis

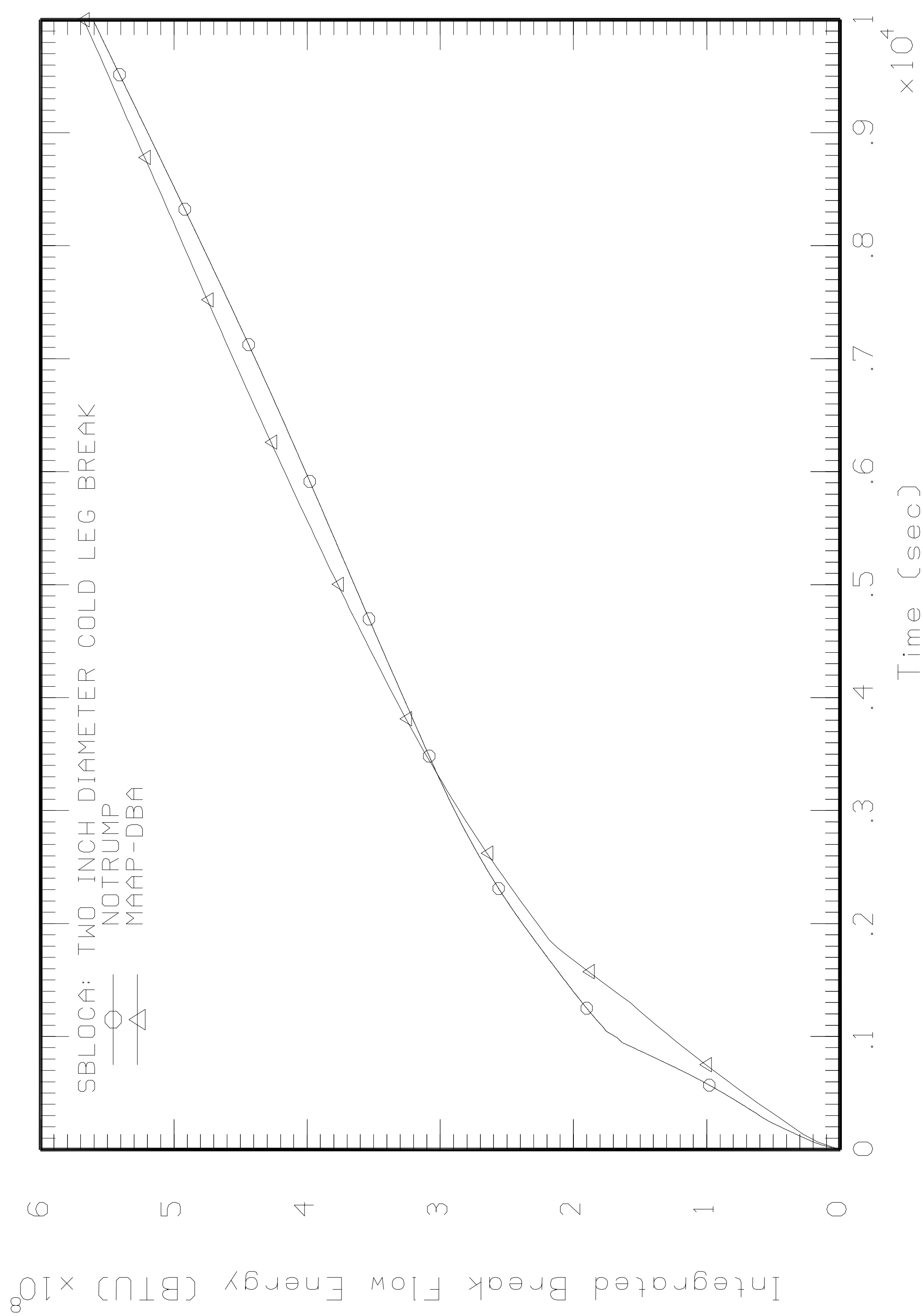


Figure 9-26b
Comparison of 2 Inch Cold Leg Releases for SBLOCA Analysis

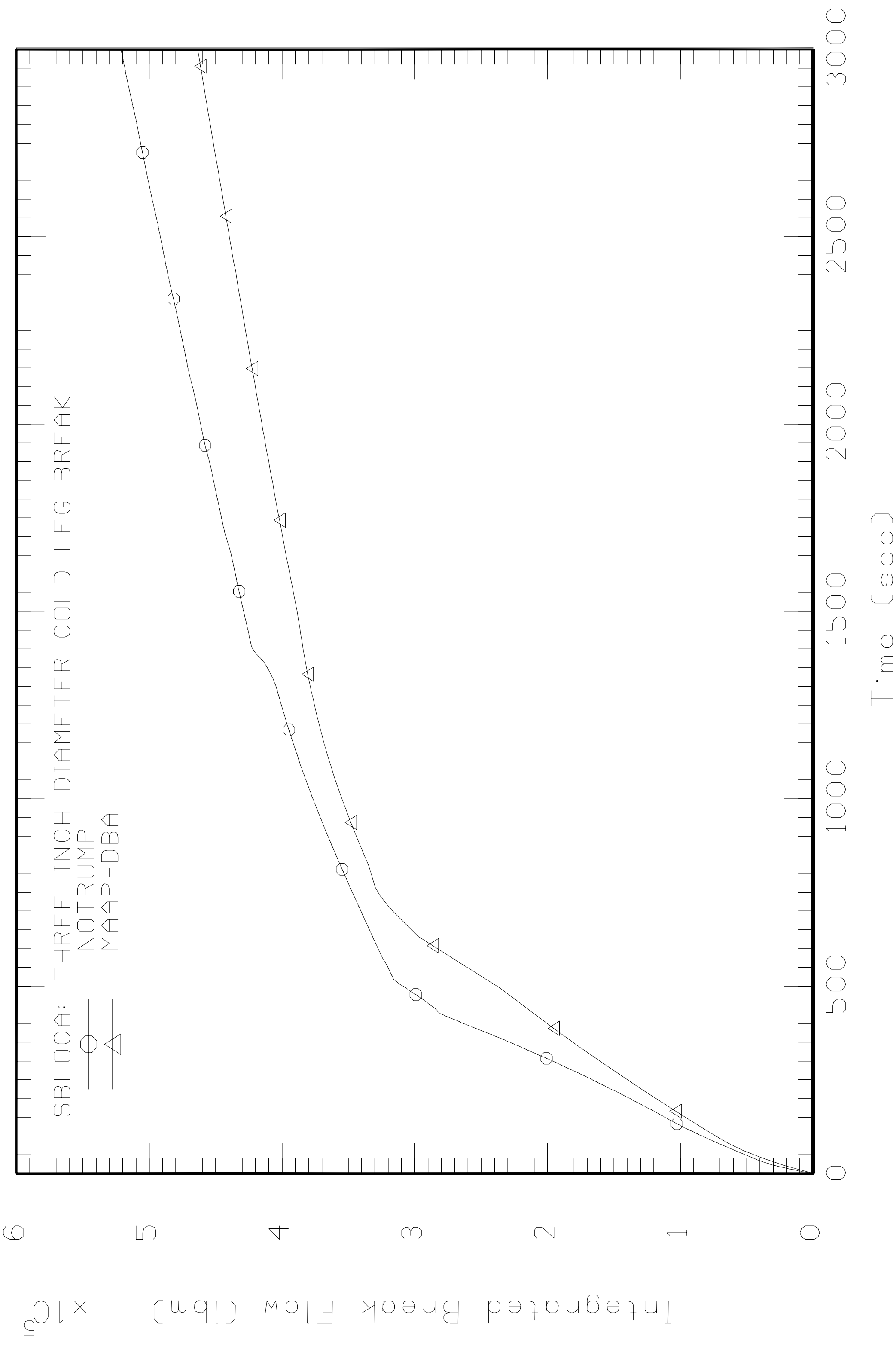


Figure 9-27a
Comparison of 3 Inch Cold Leg Releases for SBLOCA Analysis

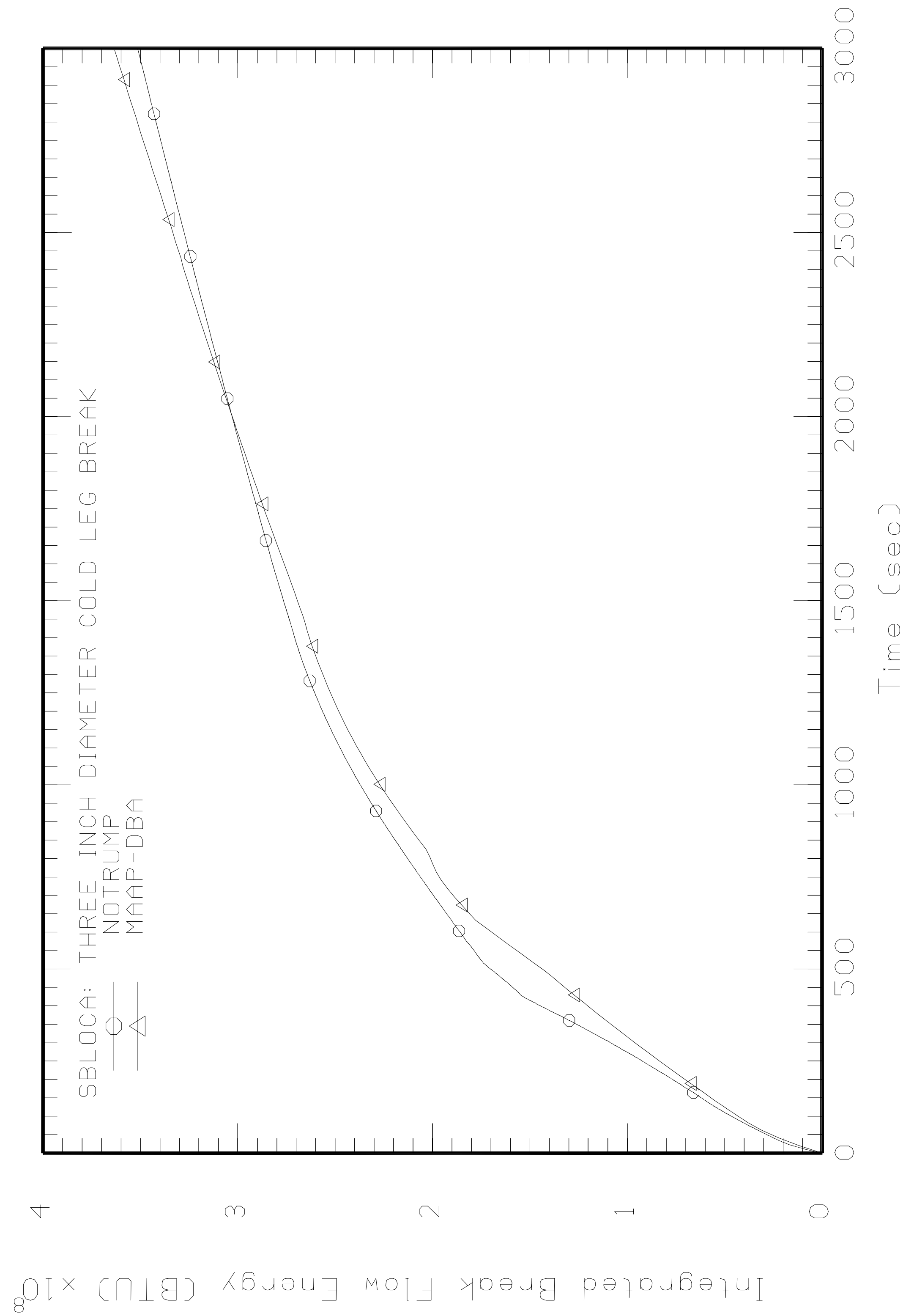


Figure 9-27b
Comparison of 3 Inch Cold Leg Releases for SBLOCA Analysis

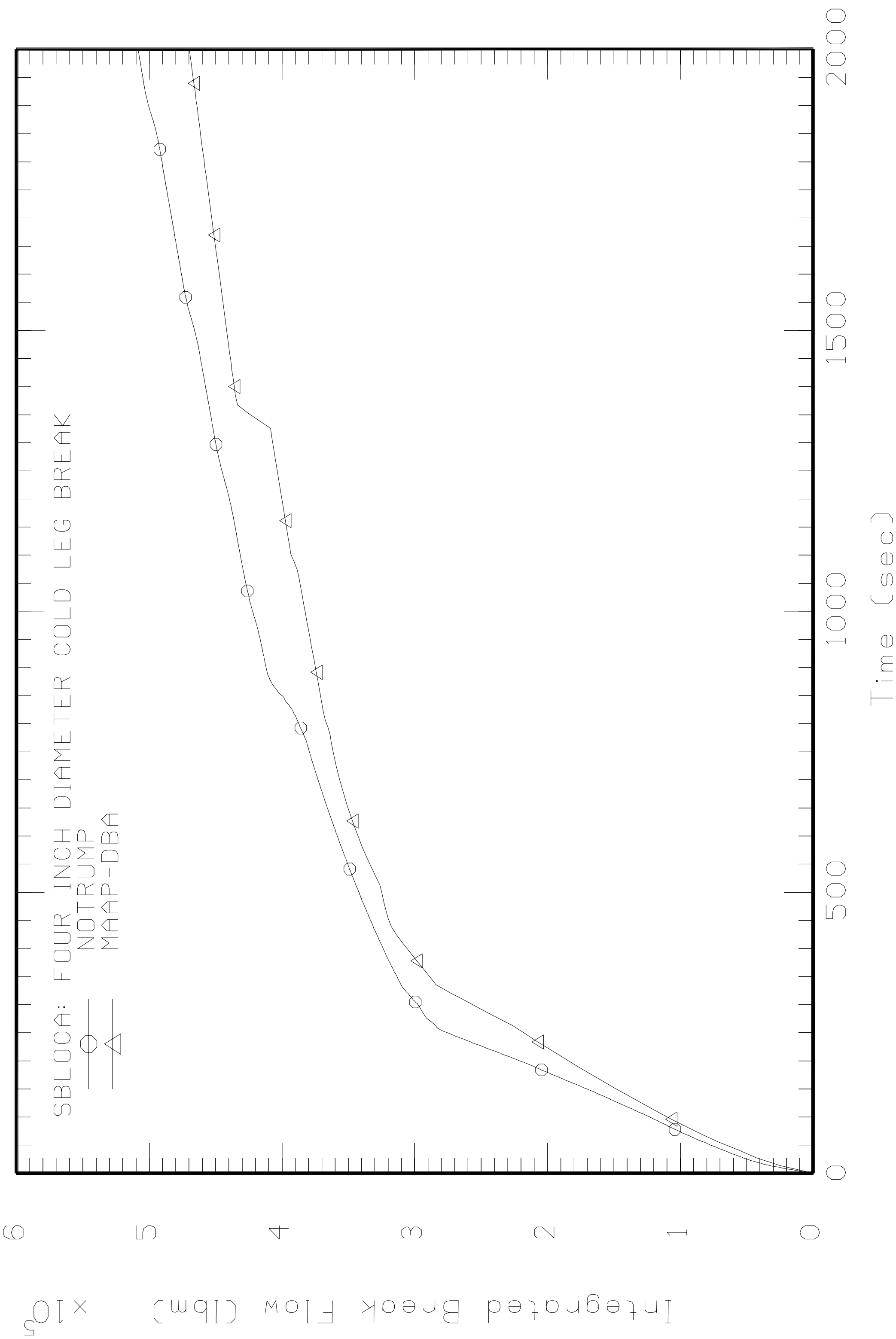


Figure 9-28a
Comparison of 4 Inch Cold Leg Releases for SBLOCA Analysis

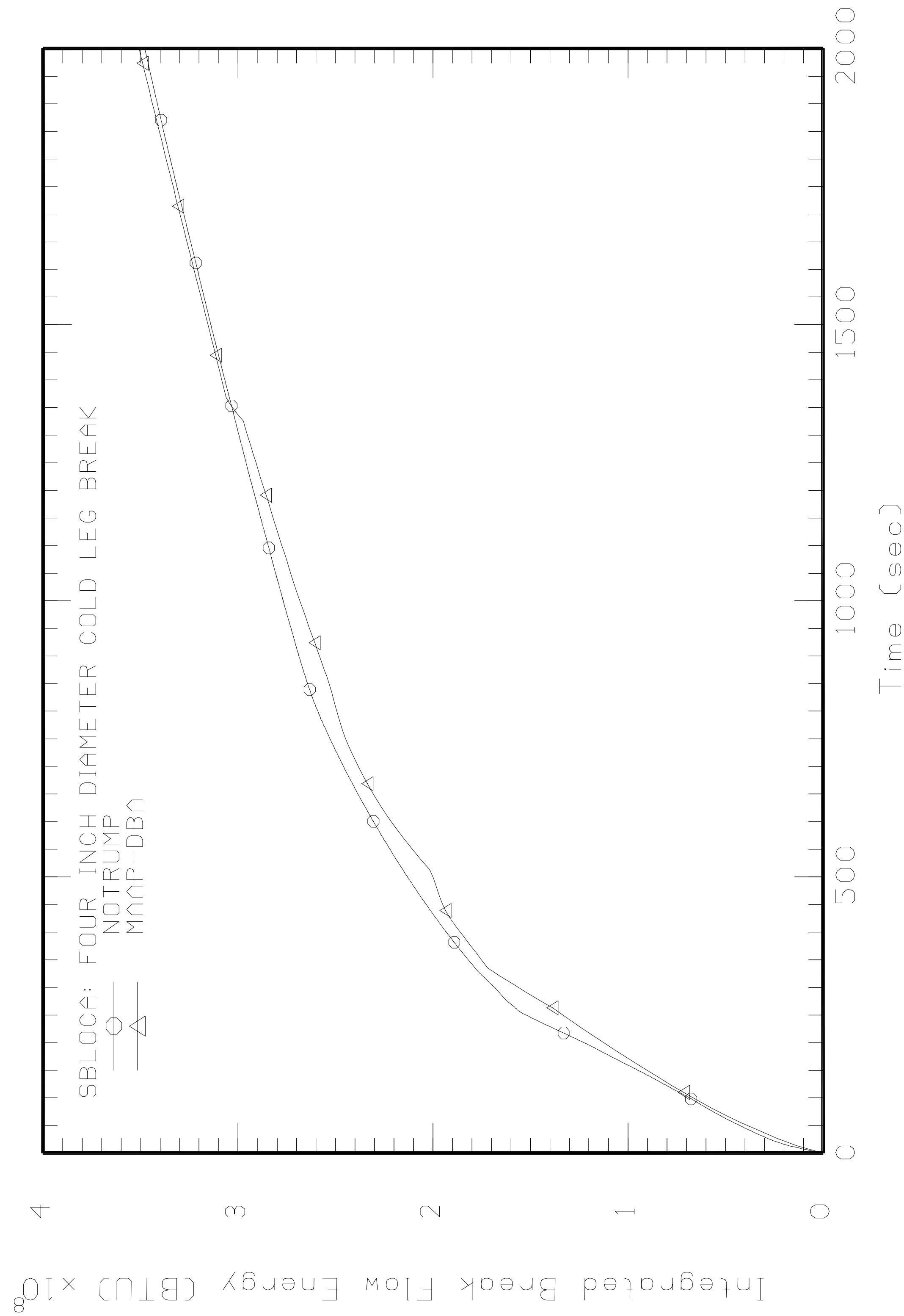


Figure 9-28b
Comparison of 4 Inch Cold Leg Releases for SBLOCA Analysis

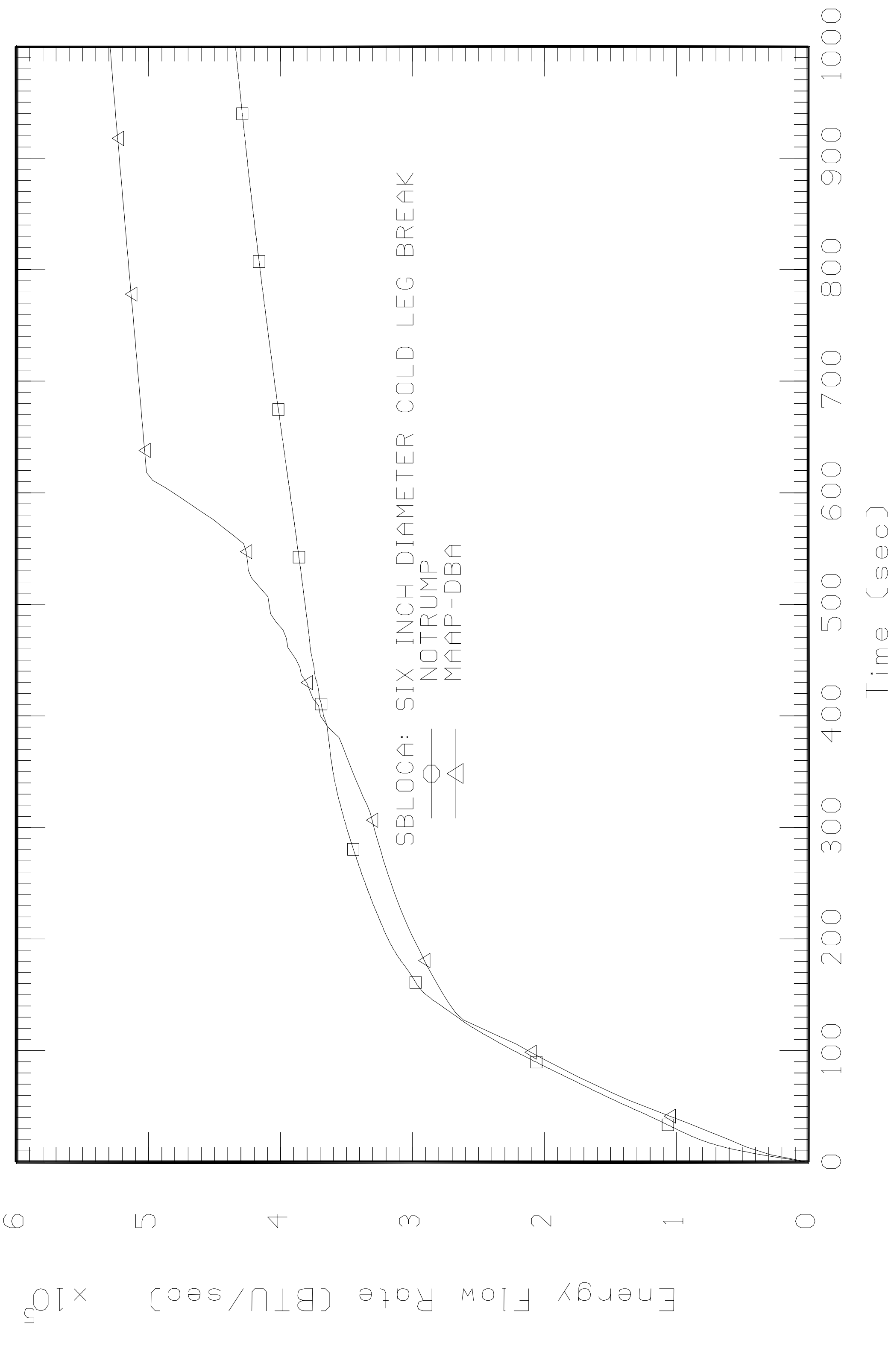


Figure 9-29a
Comparison of 6 Inch Cold Leg Releases for SBLOCA Analysis

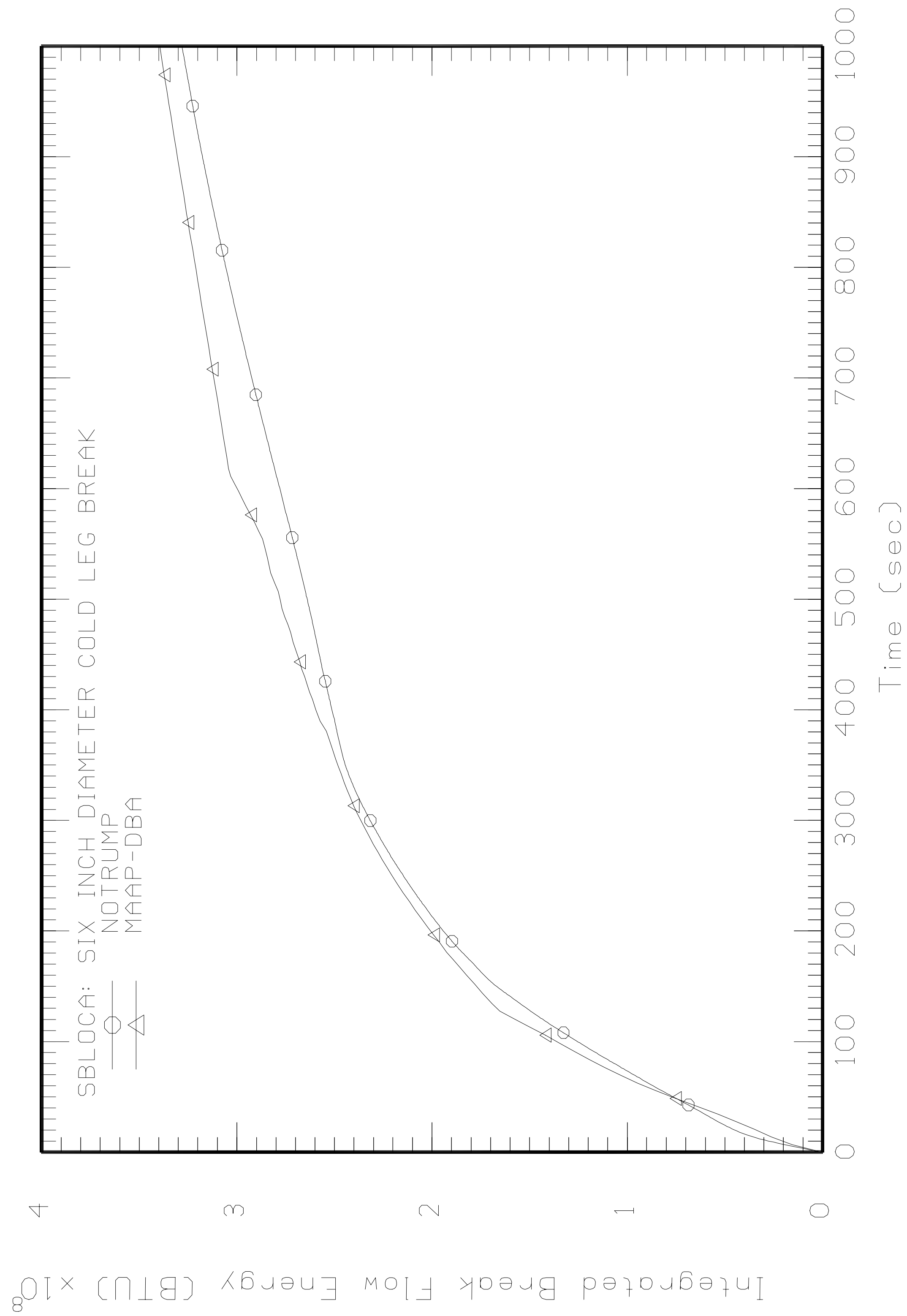


Figure 9-29b
Comparison of 6 Inch Cold Leg Releases for SBLOCA Analysis

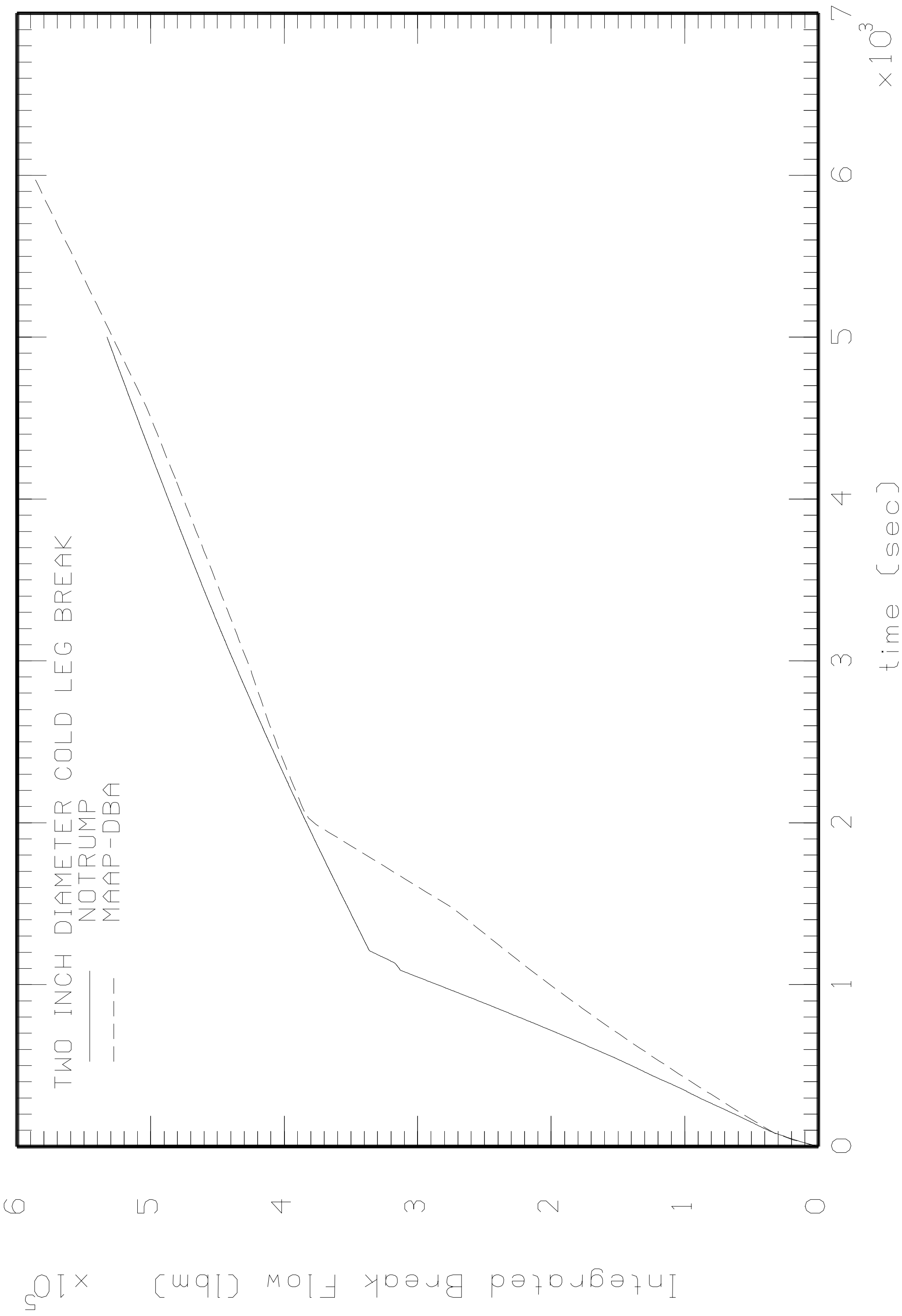


Figure 9-30a
Comparison of 2 Inch Cold Leg Releases

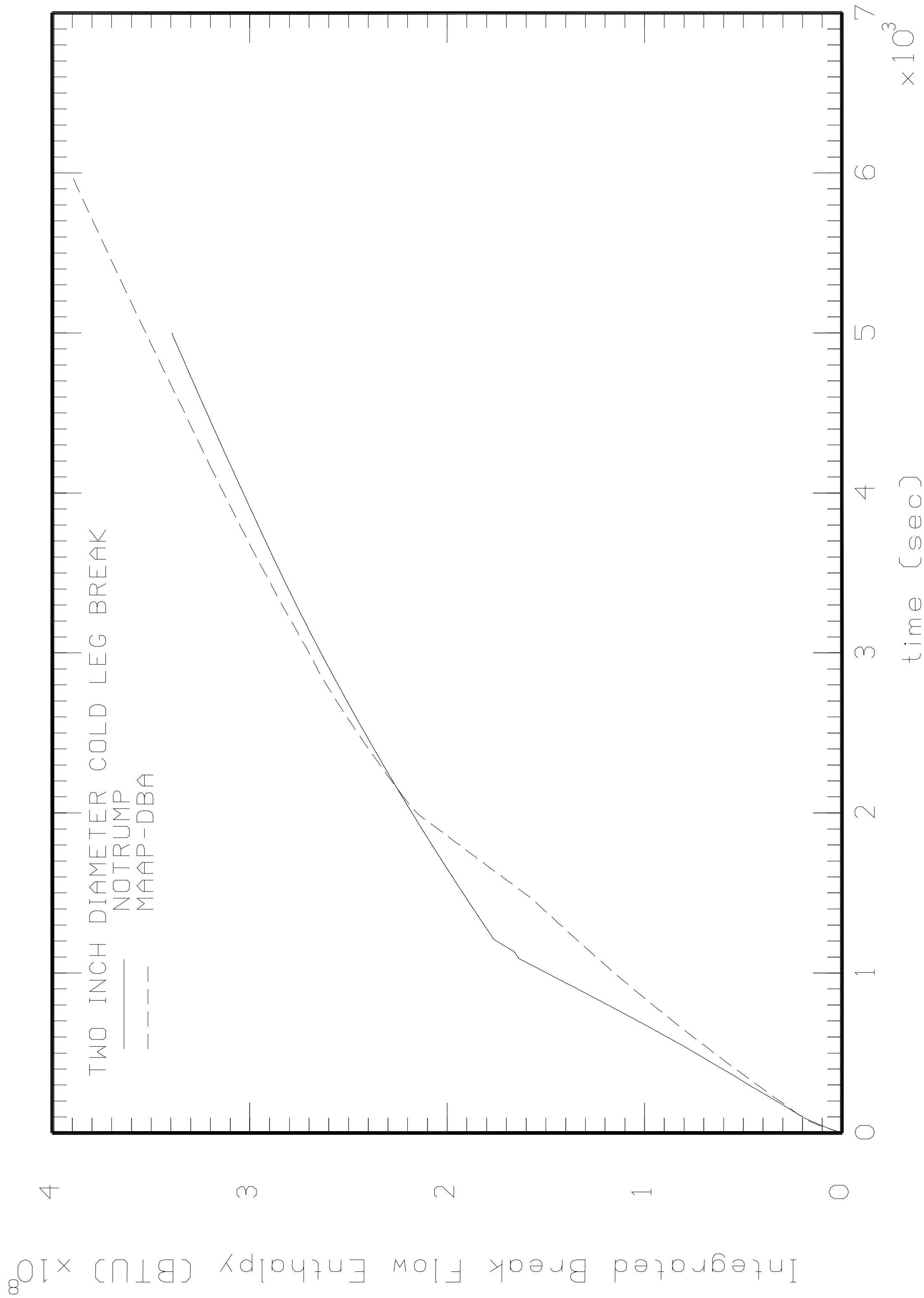


Figure 9-30b
Comparison of 2 Inch Cold Leg Releases

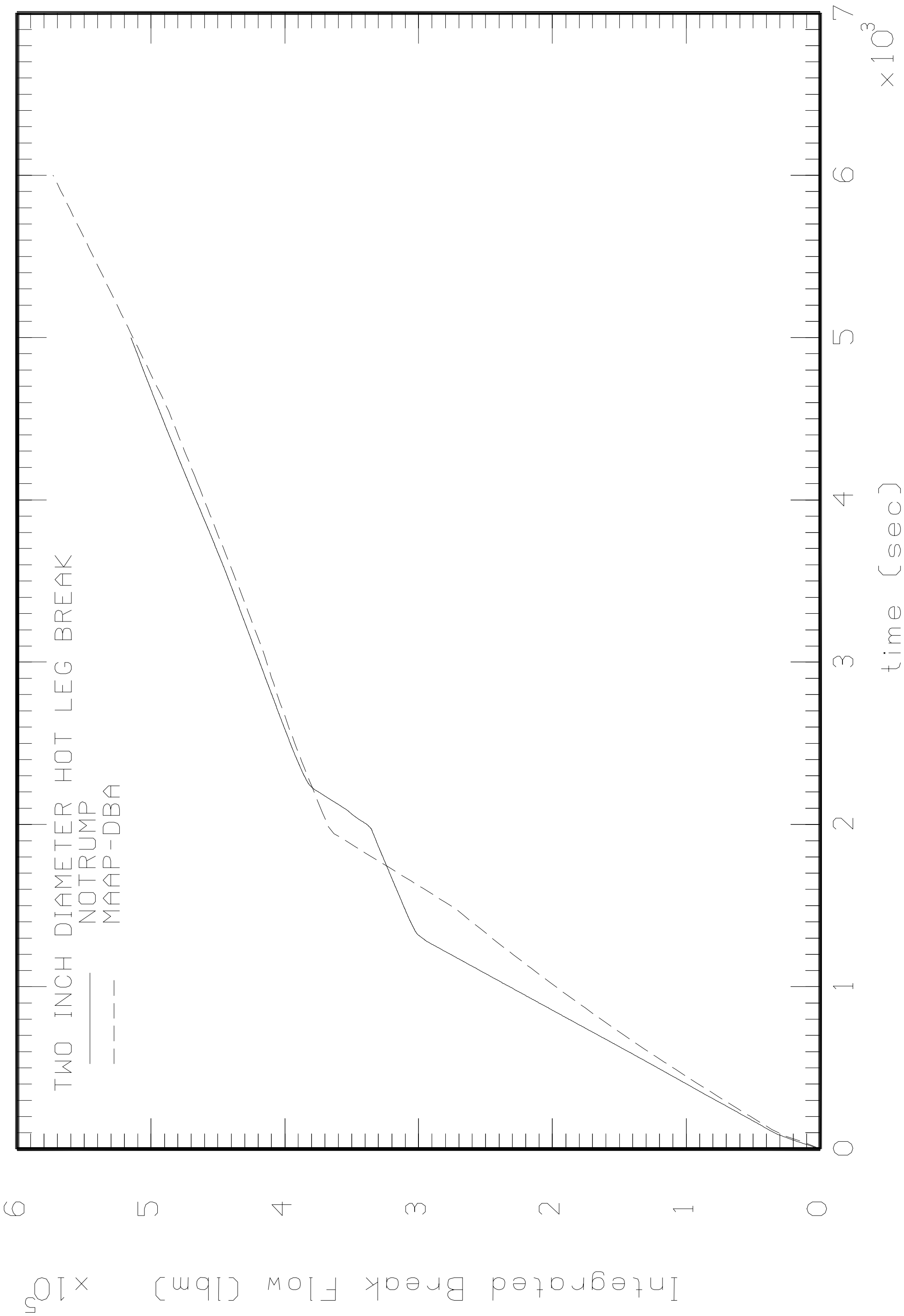


Figure 9-31a
Comparison of 2 Inch Hot Leg Releases

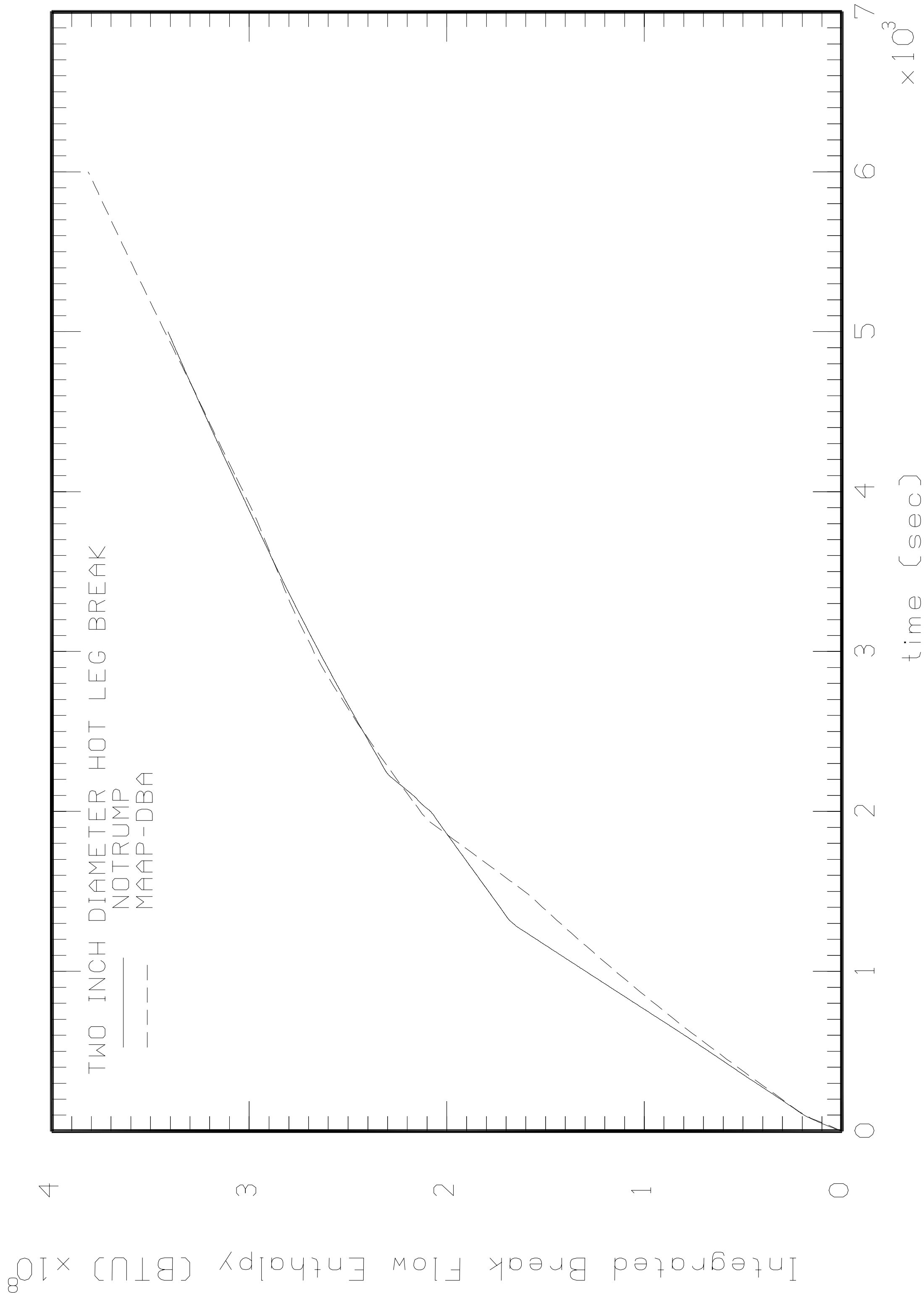


Figure 9-31b
Comparison of 2 Inch Hot Leg Releases