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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. 300 and 172

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 technical specifications. The proposed changes will revise the Operating Licenses to permit each unit to be operated with an atmospheric containment. Converting from a sub-atmospheric containment to an atmospheric containment will:

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

(b) reduce the potential for personnel injury when entering the containment during plant operations;

(c) allow for safer and more frequent containment entries for inspections and monitoring of equipment inside containment; and

(d) provide margin in the large break loss of coolant accident calculated peak clad temperature needed to support the planned future power uprate.

This license amendment request (LAR) is being provided on CD-ROM as a Portable Document Format (PDF) file. The LAR has two enclosures. Enclosure 1 is the FENOC evaluation of the proposed changes. This enclosure has five attachments. The proposed Technical Specification changes 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 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 integrity and radiological analyses conducted to support the proposed changes.

Listed in Attachment D of Enclosure 1 is a commitment where FENOC will provide, under a separate submittal, marked-up Updated Final Safety Analysis Report pages reflecting the necessary changes associated with the conversion to an atmospheric containment. Attachment D of Enclosure 1 also lists a commitment where FENOC will provide, under a separate submittal, an implementation schedule and details concerning our containment conversion implementation plan. The implementation plan will specify an implementation period for each unit.

The Beaver Valley review committees have reviewed this change. The change was determined to be safe and does 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 July 15, 2003 to support implementation following the next scheduled refueling outage for Unit 2, i.e., 2R10. Once approved, the amendments shall be implemented within the implementation periods specified in the forthcoming implementation plan.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 5, 2002.

Sincerely,

M. P. Pearson

Enclosure:

CD-ROM

c: Mr. D. S. Collins, Project Manager
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
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ENCLOSURE 1
FENOC Evaluation of the Proposed Changes

Beaver Valley Power Station
License Amendment Requests
300 (Unit 1) And 172 (Unit 2)

Subject: Application for Amendment to the Technical Specifications listed in Section 2 to permit Beaver Valley Power Station Units 1 and 2 Operation with Atmospheric Containments.

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A-1	Proposed Unit 1 Technical Specification Changes
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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. The proposed changes reflect revised containment integrity and radiological analyses, as documented in Enclosure 2, "Beaver Valley Containment Conversion Licensing Report", dated May 2002.

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 (BVPS) 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.

Changes to the following Technical Specifications (TS) are being proposed to allow each unit's containment to be operated within an atmospheric

pressure range. The proposed changes also reflect the results of a revised radiological analysis utilizing the Alternative Source Term methodology.

Affected Technical Specifications			
Change	**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, 2.c, 3.b.3 and 4.c.
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.
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	3.6.2.2	3.6.2.2	Containment Recirculation Spray System
6	3.6.2.3	N/A	Chemical Addition System
7	3.6.3.1	3.6.3.1	Containment Isolation Valves
8	N/A	3.6.4.2	Electric Hydrogen Recombiners
9a	3.7.7.1	N/A	Control Room Emergency Habitability Systems
	N/A	3.7.7	Control Room Emergency Air Cleanup and Pressurization System
9b	3.3.3.1 Table 4.3-3	N/A	Radiation Monitoring

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

Change No. 1a

Technical Specification 3.3.2.1, Engineered Safety Features Actuation System Instrumentation, revises 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. These changes are based on the new safety analysis limits for the applicable functions used in the revised containment analyses. Changes to the associated nominal setpoint values for the functions contained in the BVPS Licensing Requirements Manuals for Unit 1 and Unit 2 are shown in Attachments C-1 and C-2.

Basis for Change No. 1a

The containment pressure high and containment pressure intermediate high-high values assumed in the revised containment analyses were raised from the current values to provide sufficient margin between the revised containment operating pressure and the containment high pressure setpoint. The containment pressure high-high value was also raised in the revised containment analyses. Both minimum and maximum values were required for this setpoint to satisfy analysis requirements for all of the events analyzed including a large break LOCA, a small break LOCA and a main steam line break (MSLB). See Section 4.1.7.2 of Enclosure 2 for further discussion on the basis for this setpoint.

The proposed changes to the allowable values are based on the revised safety analysis limits shown in Table 4.1-3 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 1) and WCAP-11366, "Westinghouse Setpoint Methodology for Protection Systems, BVPS-Unit 2" (Reference 2). These WCAPs were NRC reviewed and approved as supporting information for License Amendments 239 and 120 for Unit 1 and Unit 2 respectively (Reference 3). The approved methodology described in these WCAPs was utilized to determine the new allowable values for each unit.

Change No. 1b

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 changes to the nominal setpoint values for these functions in the BVPS Licensing Requirements Manuals are shown in Attachment C-1 and C-2.

Basis for Change No. 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 No. 1c (Unit 1 change only)

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 No. 1c (Unit 1 change only)

The current Quench Spray System design includes a restricting flow orifice in parallel with a motor operated cut back valve installed downstream of each quench spray pump to provide reduced flow for subatmospheric peak pressure control. 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 automatic quench spray flow cut back

on low RWST level will be defeated prior to implementing the approved amendment.

Change No. 2

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 42.5 psig and the Unit 2 value is revised from 44.7 psig to 43.4 psig.

Basis for Change No. 2

The proposed changes to the values for Pa are consistent with the revised calculated peak accident pressures following a design basis LOCA. See Section 4.1.7.1 of Enclosure 2 for a detailed discussion on containment pressure response and the revised calculated peak accident pressures following a design basis LOCA. See Section 7.10.1 of Enclosure 2 for a discussion of the impact of the revised Pa values on the Containment Leakage Rate Testing Program and the current measured leakage rates. Unit 1 Type B and Type C leak rate tests have been performed at slightly lower values than the revised Pa value for Unit 1 (except containment electrical penetration tests). All of the applicable Unit 1 Type B and Type C leak rate tests which do not bound the revised Pa pressure will be retested during the next Unit 1 refueling outage. These tests will be based on the new Pa. Containment penetrations excluded from Type C leak rate testing and tested with water per ASME Section XI will also be retested during the refueling outage based on the new Pa. This will ensure all containment leakage rate testing pressures bound the revised value.

Change No. 3

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 psia and ≤ 16 psia. This change eliminates the need for Figure 3.6-1. The word “primary” is also removed from the containment description.

Basis for Change No. 3

The proposed change, including the deletion of Figure 3.6-1, is consistent with the Improved Standard Technical Specifications (ISTS) Limiting Condition for Operation (LCO) for an atmospheric containment design. The revised containment analyses assumptions for initial containment air pressure are ≥ 12 psia and ≤ 16 psia. The proposed containment total air pressure range will ensure that the containment operating pressure is maintained within these analysis limits. See Section 4.1.2 of Enclosure 2 for input parameters and assumptions used in the containment analysis. Deletion of the word "primary" from the containment description is an administrative change made to conform to the presentation in the ISTS.

Change No. 4

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 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 containment description.

Basis for Change No. 4

This change revises the current minimum air temperature requirements, which were based on satisfying the containment depressurization time analysis for a subatmospheric 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 No. 5

Technical Specification 3.6.2.2 for the containment recirculation spray system is revised by reducing the required recirculation spray heat

exchangers minimum river/service water flow in surveillance requirement 4.6.2.2.e.3. For Unit 1 the minimum river water flow is changed from 8,000 gpm to 7,800 gpm. For Unit 2 the minimum service water flow is changed from 11,000 gpm to 10,600 gpm.

Basis for Change No. 5

The recirculation spray system provides a means for long term cooling utilizing the river water as the ultimate heat sink. The recirculation spray system is designed to recirculate water from the containment sump through the recirculation spray heat exchangers where it is cooled by the river/service water system. The cooled water is then directed to the containment spray system and cycled back to the containment sump. Elimination of the requirement to return to and maintain subatmospheric conditions following a LOCA allows for a reduction in the heat removal requirements of the system. Some of this margin was used to allow for a small reduction in the required river/service water flow to these heat exchangers. The revised flow requirements are consistent with the assumed flows used in the containment analyses.

Change No. 6 (Unit 1 only)

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 cut back control valve.

Basis for Change No. 6 (Unit 1 only)

As discussed in proposed change No. 1c, the 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 cut back 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 No. 7

Technical Specification 3.6.3.1, Containment Isolation Valves, is revised to delete surveillance requirements 4.6.3.1.1.a.2 and 4.6.3.1.2.e and make minor format changes to surveillance requirement 4.6.3.1.1 as a result of these deletions. This change would remove 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 Licensing Requirements Manual 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 No. 7

The basis for surveillance requirements 4.6.3.1.1.a.2 and 4.6.3.1.2.e is to ensure that the containment isolation check valves will remain closed preventing the loss of subatmospheric 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 License Amendments No. 133 and 9 to Unit 1 and Unit 2 (Reference 4) excluded spring and weight loaded containment isolation check valves not subject to containment Type C leak rate testing from the lift pressure testing requirements of Surveillance Requirement 4.6.3.1. This proposed change would therefore remove the valve cycle and lift pressure testing requirements for the remaining check valves that are subject to Type C leakage testing. As discussed above, the basis for requiring these lift pressure tests is to ensure the containment would remain subatmospheric 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 subatmospheric conditions in the containment within one hour following a DBA will no longer be applicable. The basis for this lift 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.

The format changes to Surveillance Requirement 4.6.3.1.1 made as a result of the deletion of surveillance requirement 4.6.3.1.1.a.2 are considered administrative and do not add any additional technical changes to the Technical Specification.

Change No. 8 (Unit 2 only)

Technical Specification 3.6.4.2 for the Electric Hydrogen Recombiners is revised by changing the hydrogen recombiner system flow requirement in surveillance requirement 4.6.4.2 from 42 scfm to 50 scfm.

Basis for Change No. 8

The containment post-DBA hydrogen concentration was evaluated for the atmospheric containment conversion. This evaluation is discussed in Section 4.1.7.7 of Enclosure 2. The hydrogen concentration analysis increased the assumed recombiner flow rate from 42 scfm to 50 scfm in Unit 2.

Change No. 9a

Technical Specifications 3.7.7.1 (Unit 1) and 3.7.7 (Unit 2) for the Control Room Emergency Habitability Systems are revised to include the following changes:

(a) The operability requirements for the control room bottled air pressurization system are removed from both the Unit 1 and Unit 2 Technical Specifications. Specifically, LCO item b is deleted (and the associated footnote) along with the action statements and surveillance requirements related to operability of the control room bottled air system.

(b) Surveillance Requirements 4.7.7.1.1.c.1 and 4.7.7.1.2.c.1 for control room emergency habitability systems are revised by changing the Unit 1 and Unit 2 emergency ventilation filtration system's penetration. The by-pass leakage testing requirements is changed from < 1% to < 0.05% for the HEPA and charcoal filters in the Unit 1 Technical Specifications.

(c) The control room emergency ventilation system 60 minute start time delay in Surveillance Requirement 4.7.7.1.2.d.2 (Unit 1) and 4.7.7.1.e.2 (Unit 2) is deleted.

Basis for Change No. 9a

BVPS has a common control room pressure envelope for Unit 1 and Unit 2. The control room habitability systems consist of a bottled air pressurization subsystem and three emergency ventilation subsystems. The three emergency ventilation subsystems consist of one manually started Unit 1 pressurization fan and filter subsystem, and two redundant automatically started Unit 2 pressurization fans and filter subsystems. Upon receipt of a control room high radiation or containment isolation phase B (CIB) signal from either unit, the control room ventilation system is isolated from the outside air and the bottled air system is initiated to pressurize the control room for 60 minutes. After 60 minutes a Unit 2 emergency ventilation pressurization subsystem is automatically started, or the Unit 1 emergency ventilation subsystem can be manually started to pressurize the control room. Two out of three emergency ventilation subsystems are required to be operable to satisfy Technical Specification 3.7.7.1 for Unit 1. Both of the Unit 2 emergency ventilation subsystems are required to be operable to satisfy Technical Specification 3.7.7 for Unit 2.

The control room habitability dose analysis performed in support of the proposed change to an atmospheric containment is described in Chapter 5 of Enclosure 2. The analysis was performed, using selective implementation of the Alternative Source Term methodology, for a design basis LOCA and control rod ejection accident.

Several changes to the control room habitability systems have been modeled into this analysis. These changes include; elimination of the accident mitigation function of the bottled air pressurization system, elimination of the 60 minute time delay associated with initiation of the control room emergency ventilation system, an increased HEPA and charcoal filter efficiency for the Unit 1 control room emergency ventilation system to agree with the Unit 2 filter efficiencies. With the proposed changes, the automatic initiation of the bottled air system will be defeated. Upon receipt of a CIB or control room high radiation signal, the control room ventilation would be isolated and the Unit 2 emergency ventilation system automatically started after a short time delay of approximately 90 seconds.

The control room habitability analysis was performed to bound an event at either unit and therefore no credit was taken for automatic initiation of the Unit 2 emergency ventilation system. The analysis assumed that operator action is necessary to initiate the control room emergency ventilation system, and that a pressurized control room is available within 30 minutes. Unit 1 currently credits manual operator action to initiate the control room emergency ventilation system for a main steamline break and LOCA following depletion of the bottled air system. While no credit is taken in the analysis for the automatic initiation of the Unit 2 control room emergency ventilation system, reducing the system time delay to approximately 90 seconds will ensure sufficient time is available to take manual operator action in the event that neither Unit 2 pressurization subsystems are automatically started.

As discussed above, the control room habitability dose analysis in support of the containment conversion was performed for the two impacted design basis events, i.e., LOCA and a control rod ejection accident using AST methodology. A radiological assessment of the impact on control room habitability for the proposed changes on the remaining design basis events was also performed and is included in Chapter 5 of Enclosure 2. This assessment was performed using existing licensing basis methodology/assumptions/regulatory acceptance criteria, and is not included in the selected application of AST methodology. The assessment demonstrates that the control room doses for the remaining design basis events will continue to remain within the regulatory limits with the proposed changes.

The control room habitability systems requirements are applicable during movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies in addition to the applicability requirements in Modes 1 through 4. For BVPS-2 these requirements are only applicable during movement of recently irradiated fuel assemblies or movement of fuel assemblies over recently irradiated fuel assemblies. The impact of the proposed changes to the control room habitability systems on these applicability requirements is discussed below.

For BVPS-1 the control room habitability systems are not required to mitigate the consequences of a Fuel Handling Accident (FHA) in order to maintain the control room operator radiation dose to within the limits of 10CFR50.67. The FHA analysis only assumes that a 30 minute purge of the

control room atmosphere occurs following radioactivity release termination. The basis for the control room habitability systems operability requirements during movement of fuel assemblies is only to ensure that a sufficient level of control room ventilation equipment is available to conduct the 30 minute purge following a FHA. The proposed changes to the control room habitability systems will not impact this capability.

For BVPS-2 the control room habitability systems are not required for FHA mitigation unless recently irradiated fuel is being moved or fuel assemblies are being moved over recently irradiated fuel assemblies. Recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 100 hours. BVPS-2 T/S LCO 3.9.3 prohibits movement of recently irradiated fuel. Therefore no control room habitability systems are currently required for FHA mitigation. The proposed changes to the control room habitability systems will therefore have no impact on FHA mitigation.

Change No. 9b (Unit 1 only)

Technical Specification 3.3.3.1 is revised by removing reference to the control room emergency bottled air pressurization system (CREBAPS) in surveillance requirements Table 4.3-3, footnote ###, for the control room area radiation monitors.

Basis for Change No. 9b

This change is consistent with the proposed change discussed above for removing the control room bottled air pressurization system from the Technical Specifications.

3.0 BACKGROUND

BVPS Units 1 and 2 containments are subatmospheric type containments. The containments are maintained at a subatmospheric pressure (air partial pressure between 8.9 psia for Unit 1 and 9.0 for Unit 2 psia, 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 subatmospheric 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 would be revised to maintain the containments within a total air pressure range of 12 psia and 16 psia during Modes 1 through 4.

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

(a) eliminating 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 future station power uprate plans. FENOC is currently in the process of performing the station reviews and analyses for a power uprate submittal. The atmospheric containment conversion also provides margin in the Large Break LOCA calculated peak clad temperature (PCT) needed to support the planned future power uprate.

Existing Design Basis

The current subatmospheric 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.

The containment air partial pressure is maintained as a function of refueling water storage tank temperature and 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 subatmospheric conditions. Controlling containment partial pressure within the established

limits also prevents the containment pressure from violating the containment design minimum pressure of 8.0 psia in the event of an inadvertent actuation of the Quench Spray System.

The limiting Design Basis Accidents (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 design basis 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.

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.

The containment liner design temperature for both units is 280°F. 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 phase, and Westinghouse methodologies for the reflood and post-reflood phases. 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/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 current BVPS radiological analyses are based on the subatmospheric design of Units 1 and 2, where all containment leakage is assumed to terminate within 1 hour. The design basis events that result in pressure transients in the containment are the LOCA, the MSLB, and the Control Rod Ejection Accident (CREA). For a MSLB, the dose consequences of a break

inside containment are bounded by a MSLB outside of containment. The radiological analyses for these events, which are based on Total Integrated Dose (TID) 14844 and NUREG 0800 methodology, demonstrate compliance with the dose limits of 10CFR100.11 at the site boundary (EAB and LPZ), and Criterion 19 for the control room.

Proposed Changes to Design Basis

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

1. The existing containment design analysis criteria is changed to no longer require the containment to depressurize to subatmospheric conditions within 60 minutes following a design basis accident and maintain the containment subatmospheric 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 design basis LOCA within 24 hours after the postulated accident.
2. The new containment integrity analysis and supporting analysis performed for the proposed change to an atmospheric containment include the following design basis methodology changes:
   (a) The Containment Integrity Analysis calculation methodology used was changed from the LOCTIC computer code to the MAAP5 code.
   (b) The Unit 1 LOCA Mass and Energy Release calculation methodology used was changed from LOCTIC to Westinghouse approved methodology as described in WCAP-10325-P-A (Reference 5).
   (c) The Main Steamline Break Mass and Energy Release calculation methodology used was changed from the Westinghouse MARVEL code to the LOFTRAN code as described in WCAP-8822, Supplement 1 (Reference 6).
   (d) Selective implementation of the Alternative Source Term methodology was used in accordance with Regulatory Guide 1.183 (Reference 7) for dose analyses impacted by the containment conversion.

(e) The radiological analysis utilized updated control room atmospheric dispersion factors using ARCON96 methodology.

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 the License Amendment Request.

4. As a result of the revised analyses in support of the containment conversion, the accident mitigation function of some plant design features has been eliminated. These are the control room bottled air pressurization system and the Unit 1 Quench Spray flow reduction scheme.

Other Design Basis Considerations

The filtration requirements of the Supplementary Leak Collection and Release System (SLCRS) were considered in the revised radiological analyses conducted to support the conversion to an atmospheric containment. The filters were not assumed operational for the LOCA dose analyses but were assumed operational for vital access doses and equipment qualification requirements. These are conservative assumptions that support the position that operation of the SLCRS filters is not required for conversion to an atmospheric containment. However, the operability requirements of the SLCRS filters are being retained. Changes to the operability requirements of the SLCRS filters may be submitted at a later date should they prove to be advantageous to plant operations.

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 performed under the 10CFR50.59, Safety Evaluation process. Included with the description of the modification is a statement of the benefit gained by the modification. These plant modifications apply only to BVPS Unit 1.

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 future uprated power 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 also being installed to limit the mass and energy releases into the containment from a MSLB at the future uprated power level, and to support the mass and energy release analysis performed for the containment conversion. Changes to the BVPS LRM to include response time requirements for these valves are shown in Attachment C-1.

4.0 TECHNICAL ANALYSIS

The Beaver Valley Power Station (BVPS) units are currently licensed with containments that are maintained at subatmospheric pressure during normal operation. These License Amendment Requests (LAR) propose revising the BVPS Unit 1 and Unit 2 Technical Specifications such that each unit may be operated as atmospheric containments. 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/energy releases, a revised containment integrity analyses, and adoption of the Alternative Source Term methodology.

It is noted that, although the revised analysis were performed at a bounding future power uprate core power level of 2900 MWt, this LAR is not requesting approval of the uprated power level. A separate request for increasing the licensed core power level to 2900 MWt is planned to be submitted in the future. The revised analysis therefore provide conservatism, since the units will be operated at the currently licensed core power level of 2689 MWt until the future power uprates are approved.

The requested changes are based on revised containment analyses that used the Modular Accident Analysis Program (MAAP5) Containment Analysis Code at a bounding future power uprate core power level of 2900 MWt. The supporting radiological analyses selectively implement the Alternative Source Term methodology as described in Regulatory Guide 1.183 “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors” (Reference 7) for dose analysis impacted by the containment conversion.

The methodology described in WCAP-10325-P-A (Reference 5) was used to determine the new mass/energy releases for a LOCA. The methodology, described in WCAP-8822, Supplement 1 (Reference 6), was used to determine the new mass/energy releases for a main steamline break.

The revised containment integrity analyses uses the MAAP5 Containment Analysis Code. This code utilizes multiple nodes to model the containment and its sub-compartments. The existing analyses of record used a single node model. The multiple node models utilized by MAAP5 produce results that are more representative of actual post-accident conditions inside the containment. The MAAP5 containment analysis methodology has recently been submitted to the NRC for review and approval in WCAP-15844, "Topical Report on the MAAP5 PWR Large Dry Containment Model" (Reference 8). The revised containment integrity analysis demonstrates that the containment pressure and temperature response following any design basis event will remain within the current containment design bases, i.e., 45 psig for containment structural design pressure and 280°F for containment liner temperature.

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.

The revised radiological analysis utilizes the Alternative Source Term methodology. This methodology was also used in the revised Fuel Handling Accident Analysis which supported the implementation of Amendment 241 for Unit 1 and Amendment 121 for Unit 2 (Reference 9). 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 and NUREG-0800, Standard Review Plan (SRP) 15.0.1 (Reference 10). The dose impact of modifying the control room habitability systems for other design basis events was also assessed. This evaluation demonstrates that the control room doses will remain within the criteria of SRP 6.4 (Reference 11).

In addition, the impact of long-term post-LOCA containment leakage resulting from atmospheric containment conversion and associated use of

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 assessed. These evaluations demonstrate there are no adverse impacts and the continued compliance with applicable regulatory criteria.

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 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 for containments that are maintained at subatmospheric pressure during normal operation. These License Amendment Requests (LAR) propose revising the BVPS Unit 1 and Unit 2 Technical Specifications such that each unit may be operated as an atmospheric containment. The proposed changes consists of the following.

(a) Revision of the allowable values for containment pressure high, intermediate high-high and high-high setpoints.

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

(c) Deletion of Refueling Water Storage Tank Level – Auto QS Flow Reduction functional unit (Unit 1 only).

(d) Revision of the peak calculated containment pressure for a LOCA (Pa).

(e) Replacement of containment air partial pressure limits with containment total air pressure limits.

(f) Revision of the containment average air temperature requirement.

(g) Revision of the river/service water flow requirement.

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

(i) Revision to the Electric Hydrogen Recombiners requirements (Unit 2 only).

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

(k) Revision to the Control Room Emergency Habitability Systems requirements.

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). All of these accidents have been previously analyzed in the Updated Final Safety Analysis Report (UFSAR) except the feedwater line break. This is not analyzed because the MSLB is most 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. Therefore a change from a subatmospheric to an atmospheric containment will not increase the probability of these accidents.

The revised containment integrity analysis demonstrates that the pressures and temperatures associated with the applicable design basis accidents identified above are within the existing containment design

limits. From a containment integrity viewpoint, the limiting design basis accidents (DBA) presently are the MSLB for Unit 1 and the LOCA for Unit 2. Following the conversion to an atmospheric containment, the limiting DBA will be the MSLB for both units.

The affects of the proposed changes on plant structures, systems and components (SSC) has been evaluated and verify that the capability of the SSCs to perform their design functions will be retained following approval of the proposed changes.

The revised radiological analysis reflects a selective application of the Alternative Source Term (AST) of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" and incorporation of the ARCON96 methodology for on-site atmospheric dispersion factors. The revised radiological analysis concludes that normal operation of the BVPS units with atmospheric containments will not impact either unit's compliance with the operator exposure limits set forth in 10CFR20, or with the public exposure limits set forth by 10CFR50, Appendix I.

For accident conditions, the proposed changes will potentially impact the reported dose consequences of the LOCA, CREA and MSLB for both BVPS units, and the locked rotor accident (LRA) for BVPS Unit 1. The radiological consequences of the remaining design bases accidents are not adversely impacted by the proposed changes.

The revised radiological analysis concludes that site boundary and control room dose consequences of the LOCA and the CREA remain within the regulatory requirements of 10CFR50.67, as supplemented by Regulatory Guide 1.183. It also concludes that the control room doses for the MSLB for both BVPS units, and LRA for BVPS Unit 1 will continue to remain within the regulatory limits provided in SRP 6.4.

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.

Changes to instrumentation setpoints, system flow rates, surveillance requirements, and the elimination of certain operability requirements will not have any affect that could create the possibility of a new or different type of accident since none of these changes would result in any changes to the manner in which the affected equipment is operated.

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 integrity analysis conducted to support the proposed changes demonstrate that the containment peak pressure and temperature following a DBA will not exceed the containment's design limits. Since the containment design limits are not exceeded, the existing margin of safety between these limits and the containment failure limits is not reduced.

The revised radiological analysis concludes that the existing dose consequence margin of safety is not significantly reduced.

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 12), 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	Yes
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 subatmospheric 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 subatmospheric pressure within one hour.

Modification

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

Criterion 19 Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including a LOCA. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Design Conformance

A control room is provided and equipped to operate the BVPS units safely under normal and accident conditions. Control room shielding and ventilation are designed to permit continuous occupancy of the control room for the duration of any accident, including a LOCA.

Modification

The design conformance description for Criterion 19 requires modification to reflect the selective implementation of the AST methodology. This methodology was used to determine the control room doses associated with a LOCA, CREA and FHA. The use of the AST methodology for a FHA was incorporated by a previously approved amendment (Reference 9). Implementation of this methodology assures compliance with 10CFR50.67, which is now contained within Criterion 19. Therefore, conformance to Criterion 19 will continue to 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 design basis 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 maintaining 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 to be prepared 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. WCAP-11419, Revision 2, December 2000 "Westinghouse Setpoint Methodology for Protection Systems, BVPS-Unit 1".
2. WCAP-11366, Revision 4, December 2000 "Westinghouse Setpoint Methodology for Protection Systems, BVPS-Unit 2".
3. NRC Issuance of Amendment letter dated July 30, 2001, Beaver Valley Power Station License Amendments 239 (Unit 1) and 120 (Unit 2).
4. NRC Issuance of Amendment letter dated December 13, 1988, Beaver Valley Power Station License Amendments 133 (Unit 1) and 9 (Unit 2).
5. WCAP-10325-P-A (Proprietary) and WCAP-10326-A (Nonproprietary), "Westinghouse LOCA Mass & Energy Release Model for Containment Design – March 1979 Version," May 1983.
6. 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.
7. Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors".
8. WCAP-15844, Revision 0, Westinghouse Proprietary Class 2, March 2002, "Topical Report on the MAAP5 PWR Large Dry Containment Model".
9. NRC Issuance of Amendment letter dated August 30, 2001, Beaver Valley Power Station License Amendments 241 (Unit 1) and 121 (Unit 2).
10. NUREG-0800, Standard Review Plan 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms", Revision 0.

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

12. 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. 300

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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~~ 5.3 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	≥ 24.1 and ≤ 24.7~~≤ 8.3~~ 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	≥ 24.1 and ≤ 24.7~~≤ 8.3~~ 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~~ 7.3 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

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Deleted				
b. Containment				
i. Purge & Exhaust Isolation (RMVS 104 A & B)	S	R	M	**
ii. Area (RM-RM-219 A & B)	S	R	M	1,2,3,& 4
c. Control Room Isolation (RM-RM-218 A & B)	S	R	M###	1,2,3,4, and ##
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (RM 215B)	S	R#	M	1,2,3 & 4
ii. Particulate Activity RCS Leakage Detection (RM 215A)	S	R#	M	1,2,3 & 4
b. Deleted				

** During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

Surveillance interval may be extended to the upcoming refueling outage if the interval between refueling outages is greater than 18 months.

During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies.

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

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~~ 42.5 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~~ 42.5 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~~ 42.5 psig), and verify the overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq P_a$ (~~40.0~~ 42.5 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.

---

(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.

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 psia and ≤16 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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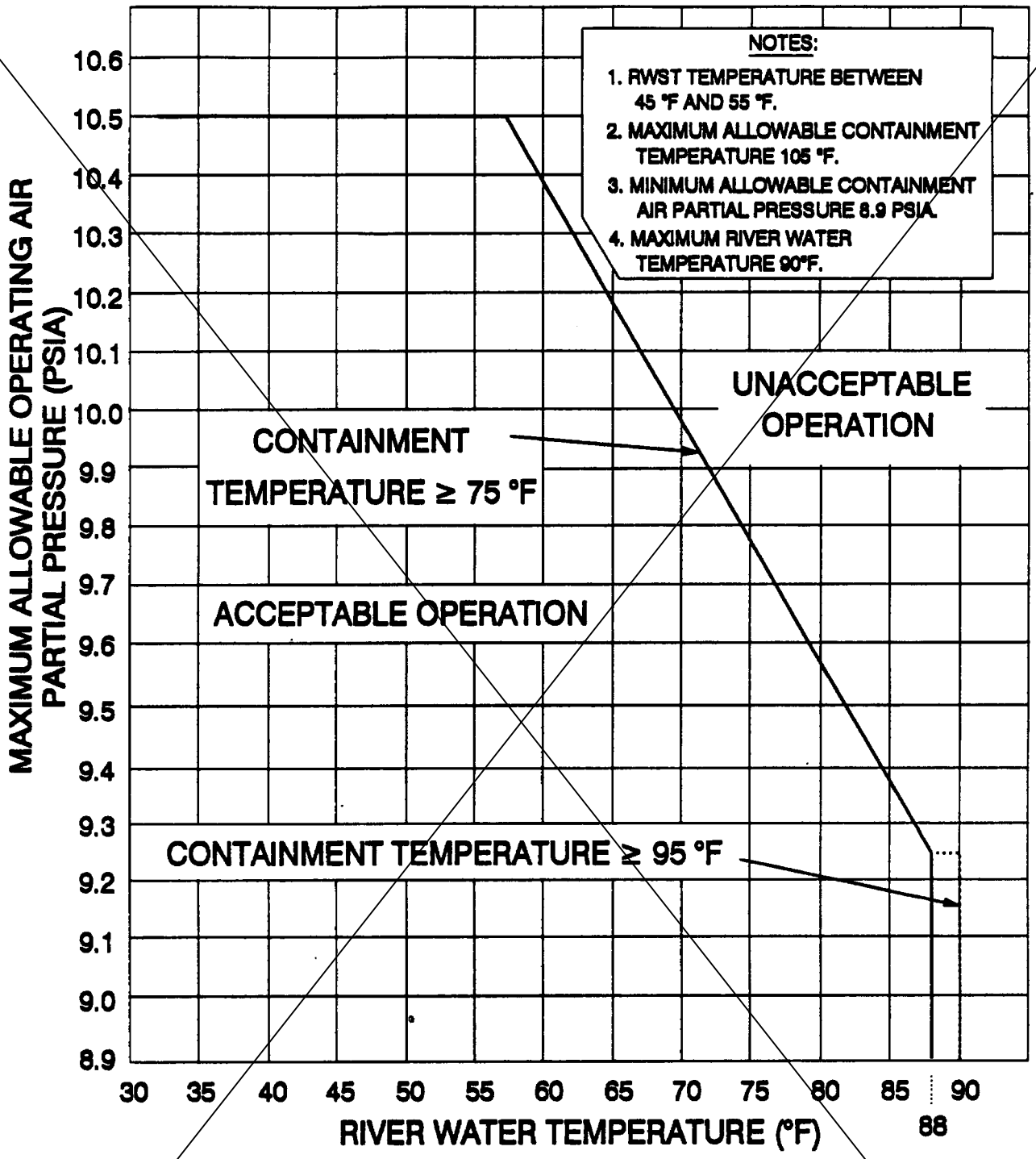

~~FIGURE 3.6-1~~

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

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 ~~Primary~~ ~~c~~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 limit 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 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"~~

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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).~~

This page is included for information/readability only.

CONTAINMENT RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Four separate and independent containment recirculation spray subsystems, each composed of a spray pump, associated heat exchanger and flow path shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment recirculation spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With two containment recirculation spray subsystems inoperable restore at least one inoperable subsystem to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each containment recirculation spray subsystem shall be demonstrated OPERABLE:

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

b. When tested pursuant to Specification 4.0.5, manually start each recirculation spray pump and verify the pump shaft rotates;

c. At least once per 18 months by verifying that on a Containment Pressure-High-High signal, the recirculation spray pumps start automatically as follows:

RS-P-1A and RS-P-2B	210 ± 5 second delay
RS-P-2A and RS-P-1B	225 ± 5 second delay

SURVEILLANCE REQUIREMENTS (Continued)

d. Verify, at the frequency specified in the Inservice Testing Program, that each recirculation spray pump's developed head at the flow test point is greater than or equal to the required developed head as specified in the Inservice Testing Program and the Containment Integrity Safety Analysis.

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

1. Cycling each power operated (excluding automatic) valve in the flow path 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. Initiating flow through each River Water subsystem and its two associated recirculation spray heat exchangers, and verifying a flow rate of at least ~~8000~~ 7800 gpm.

f. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

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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.~~

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, either:

a. Restore the inoperable valve(s) to OPERABLE* status within 4 hours, or

b. Isolate the affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 6 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 Each containment isolation valve shall be demonstrated OPERABLE*by:

~~a. At least once per 92 days by:~~

~~1~~a. Cycling, at least once per 92 days, each OPERABLE power operated or automatic valve testable during plant operation through at least one complete cycle of full travel.

---

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

~~2. 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. Immediately prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of the ~~applicable~~ cycling test, above, and verification of isolation time.

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE* during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Deleted.

d. Cycling each power operated or automatic valve through at least one complete cycle of full travel and measuring the isolation time.

~~e. 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.~~

~~f~~e. Cycling each manual valve not locked, sealed or otherwise secured in the closed position through at least one complete cycle of full travel.

---

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.1 The control room emergency habitability system is OPERABLE* when:

a. Two out of three emergency ventilation subsystems, fans, associated filters and dampers are OPERABLE, and

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

~~c~~b. The series normal air intake and exhaust isolation dampers for both units are OPERABLE, and capable of automatic closure on a CIB***~~*~~ and Control Room High Radiation isolation signal, or OPERABLE by being secured in a closed position with power removed.

~~d~~c. The control room air temperature is maintained ≤ 88°F.

APPLICABILITY: MODES 1, 2, 3 and 4, and

During movement of irradiated fuel assemblies, and

During movement of fuel assemblies over irradiated fuel assemblies.

ACTION:

a. With less than two emergency ventilation subsystems, fans, and associated filters OPERABLE, restore at least two subsystems to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

a.1 With an emergency ventilation subsystem inlet isolation damper open and not capable of being closed, the requirements of 3.0.3 are applicable.

---

* Emergency power for only one train of dampers is required in MODES 5, 6 and with no fuel assemblies in the reactor pressure vessel.

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

***~~*~~ Automatic actuation on a CIB signal is only required in MODES 1 through 4.

LIMITING CONDITION FOR OPERATION (continued)

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

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

~~c~~b. With one open series normal air intake or exhaust isolation damper inoperable and not capable of closing, restore all series dampers to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~c~~b.1 With both series normal air intake or exhaust isolation dampers inoperable and not capable of being closed, the requirements of 3.0.3 are applicable and movement of irradiated fuel assemblies and movement of fuel assemblies over irradiated fuel assemblies shall be suspended.

~~d~~c. With the control room air temperature > 88°F but ≤ 105°F, return the temperature to ≤ 88°F in 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~d~~c.1 With the control room air temperature > 105°F, be in at least HOT STANDBY within the next 4 hours and in COLD SHUTDOWN within the following 30 hours.

4.7.7.1.1 The BV-1 emergency ventilation subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is ≤ 88°F.

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

c. At least once per 18 months or after every 720 hours of system operation or (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housing or (3) following painting, fire or chemical release in any ventilation zone communicating with the system by:

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

2. Within 31 days after removal, subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying a removal efficiency of ≥ 99% for radioactive methyl iodine at an air flow velocity of .68 ft/sec with an inlet methyl iodide concentration of 1.75 mg/m^3, ≥ 70% relative humidity, and 30°C; other test conditions including test parameter tolerances shall be in accordance with ASTM D3803-1989. The carbon samples not obtained from test canisters shall be prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining a sample volume equivalent to at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Removing a longitudinal sample from an adsorber tray using a slotted-tube sampler, mixing the adsorbent thoroughly, and obtaining a sample volume equivalent to at least two inches in diameter and with length equal to the thickness of the bed.

SURVEILLANCE REQUIREMENTS (continued)

3. Verifying a system flow rate of 800 - 1000 cfm during system operation.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the ventilation system at a flow rate of 800 - 1000 cfm.

2. Verifying that on a control room high radiation/ containment phase B isolation test signal from either Unit, the system automatically closes all the series isolation ventilation system dampers which isolate the combined control room from the outside atmosphere.

3. Verifying that one emergency ventilation subsystem maintains the combined control room at a positive pressure of ≥ 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

4. Verifying that the heaters dissipate at least 3.87 kw and not exceeding 5.50 kw when tested in accordance with ANSI N510-1980.

4.7.7.1.2 The BV-2 emergency ventilation subsystems shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is ≤ 88°F.

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

c. At least once per 18 months, or after every 720 hours of system operation and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following painting, fire or chemical release in any ventilation zone communicating with the system by:

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

SURVEILLANCE REQUIREMENTS (continued)

2. Within 31 days after removal, subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying a removal efficiency of ≥ 99% for radioactive methyl iodine at an air flow velocity of 0.70 ft/sec with an inlet methyl iodine concentration of 1.75 mg/m^3, ≥ 70% relative humidity, and 30°C; other test conditions including test parameter tolerances shall be in accordance with ASTM D3803-1989. The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining a sample volume equivalent to at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Removing a longitudinal sample from an adsorber tray using a slotted-tube sampler, mixing the adsorbent thoroughly, and obtaining a sample volume equivalent to at least two inches in diameter and with a length equal to the thickness of the bed.

3. Verifying a system flow rate of 800 to 1000 cfm during system operation.

d. At least once per 18 months by:

1. Verifying that the pressure drop for the combined HEPA filters and charcoal adsorber banks is less than 5.6 inches Water Gauge while operating the ventilation system at a flow rate of 800 to 1000 cfm.

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

3. Deleted

SURVEILLANCE REQUIREMENTS (continued)

4. Verifying that one emergency ventilation subsystem maintains the control room at a positive pressure of ≥ 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

5. Verifying that the heaters dissipate at least 3.87 kw and not exceeding 5.50 kw when tested in accordance with ANSI N510-1980.

~~4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:~~

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

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

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

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

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.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is ~~40.0~~ 42.5 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.

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(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).

Containment Leakage Rate Testing Program (Continued)

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

   2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.18.b.1 & 2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

Attachment A-2

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

License Amendment Request No. 172

The following is a list of the affected pages:

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3/4 3-17
3/4 3-18
3/4 6-5a
3/4 6-6
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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.

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 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	≥ 20.8 and ≤ 21.4~~8.3~~ 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	≥ 20.8 and ≤ 21.4~~≤ 8.3~~ psig	1, 2, 3, 4	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 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.

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 ≥ Pa (~~44.7~~ 43.4 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~~ 43.4 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~~ 43.4 psig), and verify the overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a (~~44.7~~43.4 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.

---

(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.

INTERNAL PRESSURE

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 psia and ≤ 16 psia.

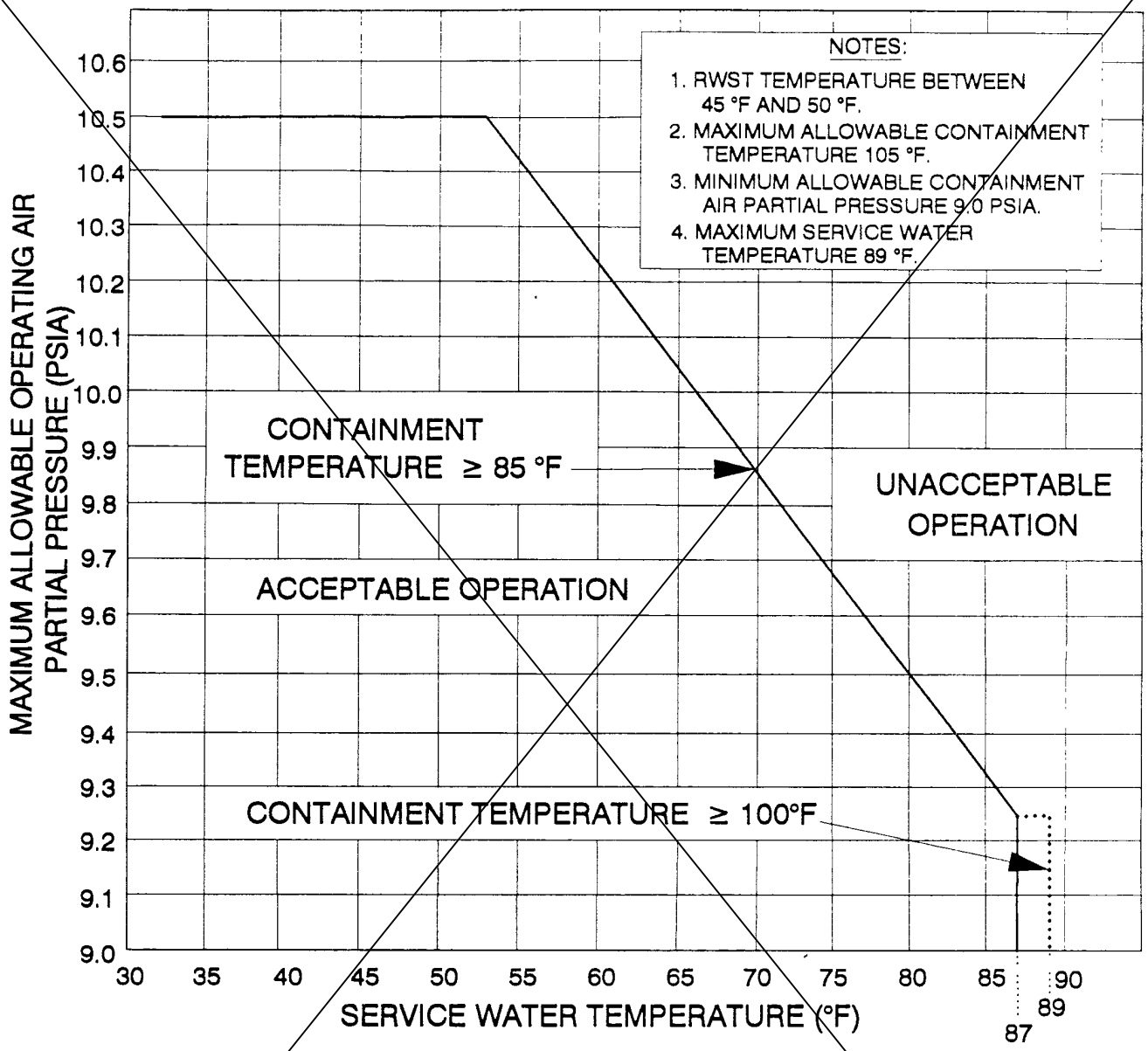
APPLICABILITY: MODES 1, 2, 3 and 4.

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.



~~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 limit 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 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"~~

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CONTAINMENT RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Four separate and independent containment recirculation spray subsystems, each composed of a spray pump, associated heat exchanger and flow path shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. For subsystems containing recirculation spray pumps 2RSS-P21A or 2RSS-P21B: With one containment recirculation spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore the inoperable spray subsystem to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

b. For subsystems containing recirculation spray pumps 2RSS-P21C or 2RSS-P21D: See action statements in Specification 3.5.2 or 3.5.3.

c. For subsystems containing recirculation spray pumps 2RSS-P21A and 2RSS-P21C, or 2RSS-P21B and 2RSS-P21D; apply Action a above if the inoperable subsystem contains recirculation spray pumps 2RSS-P21A (or B), and apply Action b above if the inoperable subsystem contains recirculation spray pumps 2RSS-P21C (or D).

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each containment recirculation spray subsystem 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. When tested pursuant to Specification 4.0.5, manually start each recirculation spray pump and verify the pump shaft rotates;

c. At least once per 18 months by verifying that on a Containment Pressure-High-High signal, each recirculation spray pump starts automatically after a 628 ± 5 second delay.

SURVEILLANCE REQUIREMENTS (Continued)

d. Verify, at the frequency specified in the Inservice Testing Program, that each recirculation spray pump's developed head at the flow test point is greater than or equal to the required developed head as specified in the Inservice Testing Program and the Containment Integrity Safety Analysis.

e. At least once per 18 months by:

1. Cycling each power operated (excluding automatic) valve in the flow path 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. Initiating flow through each Service Water subsystem and its two associated recirculation spray heat exchangers, and verifying a flow rate of at least ~~11,000~~10,600 gpm.

f. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE*.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate the affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 6 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 Each containment isolation valve shall be demonstrated OPERABLE*by:

~~a. At least once per 92 days by:~~

~~1~~a. Cycling, at least once per 92 days, each OPERABLE power operated or automatic valve testable during plant operation through at least one complete cycle of full travel.

~~2. 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. Immediately prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of the ~~applicable~~ cycling test, above, and verification of isolation time.

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE* at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Deleted.

d. Cycling each power operated or automatic valve through at least one complete cycle of full travel and measuring the isolation time pursuant to Specification 4.0.5.

~~e. 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.~~

e~~f~~. Cycling each manual valve not locked, sealed or otherwise secured in the closed position through at least one complete cycle of full travel.

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two separate and independent containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test using outside atmospheric flow rate of ≥ ~~42~~50 scfm that the heater outlet temperature increases to ≥ 700°F within 90 minutes and is maintained for at least 2 hours.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits.

2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners (i.e., loose wiring or structural connections, deposits of foreign materials, etc.).

3. Performing a hydrogen recombiner system functional test using containment atmospheric air to verify the following:

a. The recombiner blower flow when corrected to 13 psia and 130°F is ≥ ~~42~~ 50 scfm, and

b. The heater temperature increases to ≥ 1100°F within 5 hours and is maintained for at least 4 hours while operating at a recombiner blower flow that when corrected to 13 psia and 130°F is ≥ ~~42~~ 50 scfm.

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SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying the integrity of all heater electrical circuits by performing a continuity and resistance to ground test immediately following the above required functional test. The resistance to ground for any heater phase shall be ≥ 10,000 ohms.

c. Verifying that the hydrogen recombiner isolation valves (2HCS-MOV110A&B and 2HCS-MOV113A&B) are closed and de-energized after every surveillance test (per 4.6.4.2.a) is completed or after their use, post-accidents, to recombine hydrogen in the containment is completed.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP AND PRESSURIZATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 The Control Room Emergency Air Cleanup and Pressurization System comprised of the following shall be OPERABLE:

a. A pressurization filtration unit comprised of two trains of fans and filters, and flow path control dampers.*~~*~~

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

~~c~~b. Two isolation dampers in series in each of four normal air flow paths (two intake and two exhaust) with each damper OPERABLE by automatic actuation**~~*~~ or OPERABLE by being secured in a closed position with power removed.

APPLICABILITY: MODES 1, 2, 3 and 4, and

During movement of recently irradiated fuel assemblies, and

During movement of fuel assemblies over recently irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3 and 4:

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

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

~~~~ Emergency backup power for only one train of dampers and fans of the pressurization filtration unit is required in MODES 5, 6 and with no fuel assemblies in the reactor pressure vessel.

**~~*~~ Automatic actuation on a CIB signal is only required in MODES 1 through 4.

LIMITING CONDITION FOR OPERATION (continued)

ACTION (Continued)

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

a. With one train of the pressurization filtration unit ~~or one subsystem of the bottled air pressurization system,~~ or one of two isolation dampers in series inoperable, restore the inoperable system to OPERABLE status within 7 days or suspend all operations involving movement of recently irradiated fuel assemblies and movement of fuel assemblies over recently irradiated fuel assemblies.

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

SURVEILLANCE REQUIREMENTS

4.7.7.1 The Control Room Emergency Air Cleanup and Pressurization System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is ≤ 88°F.

b. At least once per 31 days by~~:~~

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

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

c. At least once per 18 months or (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings by:

1. Verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the pressurization filtration system at a flow rate of 800 to 1000 cfm.

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SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the HEPA filter banks remove ≥ 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the pressurization filtration system at a flow rate of 800 to 1000 cfm.

3. Verifying a system flow rate of 800 to 1000 cfm during system operation.

d. At least once per 18 months or (1) after 720 hours of system operation, or (2) following painting, fire or chemical release in the vicinity of control room outside air intakes while the system is operating, within 31 days after removal, subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying a removal efficiency of ≥ 99% for radioactive methyl iodide at an air flow velocity of 0.7 ft/sec with an inlet methyl iodide concentration of 1.75 mg/m^3, ≥ 70% relative humidity, and 30°C; other test conditions including test parameter tolerances shall be in accordance with ASTM D3803-1989. The carbon samples not obtained from test canisters shall be prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

e. At least once per 18 months by:

1. Verifying that the pressure drop for the combined HEPA filters and charcoal adsorber banks is less than 5.6 inches Water Gauge while operating the pressurization filtration system at a flow rate of 800 to 1000 cfm.

SURVEILLANCE REQUIREMENTS (Continued)

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

3. Deleted

4. Verifying that the pressurization filtration system maintains the control room at a positive pressure of ≥ 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

5. Verifying that the heaters dissipate at least 3.87 kw and not exceeding 5.50 kw when tested in accordance with ANSI N510-1980.

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

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

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained in accordance with the applicable record retention provision of the quality assurance program description included in the Updated Final Safety Analysis Report. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the OSC and the approval of the plant manager, predesignated alternate or a predesignated manager to whom the plant manager has assigned in writing the responsibility for review and approval of specific subjects.

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.

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(1) Exemptions to Appendix J of 10 CFR 50, as stated in the operating license.

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~~ 43.4 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. 300

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 7-5

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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.

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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~~42.5 psig following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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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 psia to 16 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.

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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 12.0 psia and 16.0 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~~ 42.5 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.

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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.

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3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEM

The OPERABILITY of the control room emergency habitability system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The ambient air temperature is controlled to prevent exceeding the allowable equipment qualification temperature for the equipment and instrumentation in the control room. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent, or 5 rem TEDE, as applicable. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50 or 10 CFR 50.67, as applicable.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

The OPERABILITY of the SLCRS provides for the filtering of postulated radioactive effluents resulting from leakage of LOSS OF COOLANT ACCIDENT (LOCA) activity from systems outside of the Reactor Containment building, such as Engineered Safeguards Features (ESF) equipment, prior to their release to the environment. This system also collects potential leakage of LOCA activity from the Reactor Containment building penetrations into the contiguous areas ventilated by the SLCRS except for the Main Steam Valve Room and Emergency Air Lock. ~~System operation was also assumed in that portion of the Design Basis Accident (DBA) LOCA analysis which addressed ESF leakage following the LOCA, however, n~~No credit for SLCRS operation was taken in the DBA LOCA analysis for collection and filtration of Reactor Containment building leakage and ESF leakage effluents ~~—~~even though an unquantifiable amount of contiguous area penetration leakage and ESF leakage effluents would in fact be collected and filtered. ~~Based on the results of the analyses, the SLCRS must be OPERABLE to ensure that ESF leakage following the postulated DBA LOCA will not exceed 10 CFR 100 limits.~~

Attachment B-2

Beaver Valley Power Station, Unit No. 2

Proposed Technical Specification Bases Changes

License Amendment Request No. 172

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 6-12
B 3/4 6-13
B 3/4 7-5

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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~~ 43.4 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.

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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 psia to 16 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.

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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 12.0 psia and 16.0 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~~ 43.4 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.

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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

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This page is included for information/readability only.

3/4.6.2.3 CHEMICAL ADDITION SYSTEM

The OPERABILITY of the chemical addition system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

The 18-month surveillance interval is consistent with expected length of fuel cycles and allows for component testing to be performed during plant shutdown conditions if necessary to avoid a plant transient that could occur if the component were tested at power. However, for those components that may be safely tested at power, the 18-month surveillance may be met by performing the required testing at power.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for both a LOCA and major secondary system breaks.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

The 18-month surveillance interval is consistent with expected length of fuel cycles and allows for component testing to be performed during plant shutdown conditions if necessary to avoid a plant transient that could occur if the component were tested at power. However, for those components that may be safely tested at power, the 18-month surveillance may be met by performing the required testing at power.

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

3/4.6.4 COMBUSTIBLE GAS CONTROL (Continued)

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 required system flow rate (~~42~~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.

The equation specified below shall be used when performing Surveillance 4.6.4.2.b.3 to correct the flow measured under test conditions to the corresponding flow at design basis post accident containment conditions of 13 psia and 130°F.

$$scfm_{PA} = scfm_{Test}\,(0.00154)\,\frac{T_C}{P_{Blower}}\left(1 + 2682.45\,\frac{P_C}{T_C^2}\right)\left(13 - 0.022\,T_C\left(1 - \frac{P_{Blower}}{P_C}\right)\right)$$

where:

T_C = average containment temperature during testing (°R)

P_{Blower} = blower suction pressure during testing (psia)

P_C = containment pressure during testing (psia)

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3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP AND PRESSURIZATION SYSTEM (Continued)

The control room air cleanup ~~system includes two~~ and pressurization system~~s. The~~consists of two redundant filtration pressurization system~~s~~ which draw~~s~~ outside air through filters and Unit 1 and Unit 2 air intake and exhaust isolation dampers. ~~The bottled air pressurization system pressurizes by discharge of air from bottles without filtration and with closure of intake and exhaust dampers. Although the bottles are shared with Unit 1, the discharge can be initiated by Unit 2 control systems in response to radiation levels.~~ Closure of the intake and exhaust dampers can be initiated by Unit 2 control systems. However, closure of dampers in one intake and in one exhaust is dependent upon availability of Unit 1 power sources.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

The OPERABILITY of the SLCRS provides for the filtering of postulated radioactive effluents resulting from leakage of loss of coolant accident (LOCA) activity from systems outside of the Reactor Containment building, such as Engineered Safeguards Features (ESF) equipment, prior to their release to the environment. This system also collects potential leakage of LOCA activity from the Reactor Containment building penetrations into the contiguous areas ventilated by the SLCRS except for the Emergency Air Lock. ~~System operation was also assumed in that portion of the Design Basis Accident (DBA) LOCA analysis which addressed ESF leakage following the LOCA, however, n~~No credit for SLCRS operation was taken in the DBA LOCA analysis for collection and filtration of Reactor Containment building leakage and ESF leakage effluents even though an unquantifiable amount of contiguous area penetration leakage and ESF leakage effluents would in fact be collected and filtered. ~~Based on the results of the analyses, the SLCRS must be OPERABLE to ensure that ESF leakage following the postulated DBA LOCA will not exceed 10 CFR 100 limits.~~

Attachment C-1

Beaver Valley Power Station, Unit No. 1

Proposed Licensing Requirements Manual Changes

License Amendment Request No. 300

The following is a list of the affected pages:

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

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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)	≤ 27.0(3)
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)

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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	≤ 85.0[7]
b. Containment Isolation-Phase "B"	Not Applicable
c. Control Room Ventilation Isolation	≤ 22.0[8]/77.0[7]

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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)	

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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~~ 24.4 psig

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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~~ 24.4 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

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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 RV-1CC-272	< 60 N/A	(B)TV-1CC-105D2	< 60
26	CCR from RCP 1A Thermal Barrier	(B)TV-1CC-107E1 RV-1CC-273	< 60 N/A	(B)TV-1CC-107E2	< 60
27	CCR from RCP 1A Motor	(B)TV-1CC-105E1 RV-1CC-274	< 60 N/A	(B)TV-1CC-105E2	< 60
28	RCS Letdown	(A)TV-1CH-200A (A)TV-1CH-200B (A)TV-1CH-200C (1)MOV-1CH-142 RV-1CH-203	< 60 < 60 < 60 N/A N/A	(A)TV-1CH-204	< 60
29	Primary Drain Transfer Pump #1 Discharge	(A)TV-1DG-108A RV-1DG-102	< 60 N/A	(A)TV-1DG-108B	< 60
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

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)
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	N/A
48	Primary Vent Header	(A)TV-1DG-109A2	< 60	(A)TV-1DG-109A1	< 60

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)
(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. 172

The following is a list of the affected pages:

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

Provided for Information Only.

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	21.1 ~~8.0~~ 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.

Provided for Information Only.

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	21.1 ~~8.0~~ 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.

Provided for Information Only.

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				

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-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

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-28	Reactor Coolant Letdown	(A)2CHS-AOV200A (A)2CHS-AOV200B (A)2CHS-AOV200C (1)2CHS-HCV142 2CHS-RV203	< 60 < 60 < 60 N/A N/A	(A)2CHS-AOV204	< 60
X-29	Pri Dr. Trans Pump Disch	(A)2DGS-AOV108A	< 60	(A)2DGS-AOV108B 2DGS-RV115	< 60 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 (2)2SIS-MOV840	N/A 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 2DAS-RV110	< 60 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-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

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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	5
		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

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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				

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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. 300 (Unit 1) and 172 (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/Corrective Actions, at Beaver Valley on (724) 682-5284 of any questions regarding this document or associated regulatory commitments.

COMMITMENT	ENCLOSURE 2 SECTION NO.	DUE DATE
1. The Emergency Operating Procedures will be provided with guidance that, for small break LOCAs, manual actions are required to initiate the containment depressurization system prior to reaching the RWST automatic transfer to recirculation setpoint.	4.1.7.2	Prior to implementation.
2. The installation and testing of cavitating venturi flow elements in the Unit 1 Auxiliary Feedwater System will be completed prior to implementation of the approved amendment.	6.3.12 6.3.16	Prior to implementation.
3. The installation and testing of new Main Feedwater Isolation Valves in Unit 1 will be completed prior to implementation of the approved amendment.	6.3.12 6.3.15	Prior to implementation.
4. A maximum linear heat generation rate limitation of 6.3 kW/ft peak rod average power for burn-ups exceeding 54 GWD/MTU will be incorporated into future BVPS core designs to support assumptions for core gap release fractions used in the CREA radiological analysis. This design constraint will be reflected in the BVPS Core Design and Fuel Cycle Implementation Administrative Procedure.	5.3.6.4	Prior to implementation.

COMMITMENT	ENCLOSURE 2 SECTION NO.	DUE DATE
5. A review of the training simulators for the containment conversion will be conducted and revalidation, in accordance with ANSI/ANS 3.5-1985, will be completed by the next regularly scheduled benchmarking following implementation.	7.1	Next scheduled simulator benchmarking following implementation.
6. FENOC will provide, under a separate submittal, markedup UFSAR pages reflecting the necessary changes associated with the conversion to an atmospheric containment.	N/A	Six months following LAR transmittal.
7. FENOC will provide, under a separate submittal, a requested implementation schedule and details concerning our containment conversion implementation plan.	N/A	Eight months following LAR transmittal.

Beaver Valley Containment Conversion Licensing Report May 2002

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 radiological analysis is done with traditional Technical Information Document (TID)-14844 source terms.

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 (MAAP5) 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.

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 separate power uprate licensing amendments reports (LAR) are 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)
ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
ANS	American Nuclear Society
AOV	Air Operated Valve
AST	Alternative Source Term
BHP	Brake Horsepower
BVPS	Beaver Valley Power Station
CAT	Chemical Addition Tank
CCW	Component Cooling Water
CEDE	Committed Effective Dose Equivalent
CFM	Cubic Feet Per Minute
CFR	Code of Federal Regulation
CGCS	Combustible Gas Control System
CH/SI	Charging/Safety Injection
CIA	Containment Isolation Phase A
CIB	Containment Isolation Phase B
CR	Control Room
CRDM	Control Rod Drive Mechanisms
CREA	Control Rod Ejection Accident
CREBAPS	Control Room Emergency Bottled Air Pressurization System
CREVS	Control Room Emergency Ventilation System
CSE	Containment System Experiment
CVTR	Carolinas and Virginia Test Reactor
DBA	Design Basis Accident
DBE	Design Basis Event
DCF	Dose Conversion Factor

LIST OF ACRONYMS (Continued)

DDE	Deep Dose Equivalent
DECL	Double Ended Cold Leg
DEHL	Double Ended Hot Leg
DEPS	Double Ended Pump Suction
DER	Double Ended Rupture
EAB	Exclusion Area Boundary
EAC	Emergency AC
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generators
EOF	Emergency Operation Facility
EQ	Equipment Qualification
ERF	Emergency Response Facility
ER-OL	Environmental Report – Operating License
FAI	Fauske Associates Incorporated
FES	Final Environmental Statement
FHA	Fuel Handling Accident
FIV	Feedwater Isolation Valve
FRV	Feedwater Regulator Valve
GDC	General Design Criteria
GSI	Generic Safety Issue
HDR	Heiss Dampf Reaktor
HEM	Homogeneous Equilibrium Model
HEPA	High Efficiency Particulate Assembly
HHSI	High Head Safety Injection
HP	Horsepower
I & C	Instrumentation and Control
IAM	Input Assumptions Methodology

LIST OF ACRONYMS (Continued)

IAR	Inputs Assumptions Requirements
ICRP	International Committee Radiation Protection
IFM	Intermediate Flow Mixer
ILRT	Integrated Leak Rate Test
IRS	Inside Recirculation Spray
LAR	License Amendment Request
LBB	Leak Before Break
LBM	Licensing Basis Methodology
LDR	Limited Displacement Rupture
LERF	Large Early Release Frequency
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
LRA	Locked Rotor Accident
M&E	Mass and Energy
MAAP5	Modular Accident Analysis Program Version 5
MAO	Maximum Allowable Operating Air
MFLB	Main Feedwater Line Break
MFW	Main Feedwater
MMF	Minimum Measure Flow
MSCV	Main Steam Check Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steamline Break
MSSV	Main Steam Safety Valve
MWT	Megawatts Thermal
NNS	Non Nuclear Safety
NPDES	National Pollutant Discharge Elimination System

LIST OF ACRONYMS (Continued)

NPSH	Net Positive Suction Head
NPSHA	Net Positive Suction Head Average
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUPPSCO	Nuclear Power Plant Standards Committee
ORS	Outside Recirculation Spray
Pa	Calculated Peak Accident Pressure
PAOT	Post Accident Operating Time
PCT	Peak Clad Temperature
PCWG	Power Capability Working Group
PERC2	Passive Evolutionary Regulatory Consequence Code
PIRT	Phenomenological Identification Ranking Table
PSIA	Pounds Square Inch Absolute
PSIG	Pounds Square Inch Gauge
PWR	Pressurized Water Reactor
QS	Quench Spray
QSS	Quench Spray System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REA	Rod Ejection Accident
RECIRC	Recirculation
RG	Regulatory Guide
RFA	Robust Fuel Assembly
RPM	Revolution Per Minute
RS	Recirculation Spray
RSHX	Recirculation Spray Heat Exchanger
RSS	Recirculation Spray System

LIST OF ACRONYMS (Continued)

RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SBO	Station Blackout
SCFM	Standard Cubic Feet Per Minute
SER	Safety Evaluation Report
SG	Steam Generator
SGTP	Steam Generator Tube Plugging
SI	Safety Injection
SIS	Safety Injection System
SJAE	Steam Jet Air Ejector
SLCRS	Supplemental Leak Collection and Release System
SSPS	Solid State Protection System
Tavg	Temperature Average
TDF	Thermal Design Flow
TDR	Time Delay Relay
TEDE	Total Effective Dose Equivalent
TID	Technical Information Document
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
VCT	Volume Control Tank

1.0 INTRODUCTION AND PURPOSE

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 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 BVPS-2 NSSS power level of 2697 MWT includes a small 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. Conversion of the containment pressure design basis from sub-atmospheric to atmospheric on BVPS-1 and BVPS-2,

2. additional large power uprating to the ultimate power level of 2910 MWT on BVPS-1 and BVPS-2, and

3. replacement of the current steam generators on BVPS-1.

Analyses for these elements of the Full Potential Program are being performed in an integrated fashion and the elements will be implemented separately. Conversion of the containment pressure design basis from sub-atmospheric to atmospheric will be implemented first in order to obtain the direct benefits such as improved personnel safety during containment entries as well as the benefit of a reduction in calculated peak clad temperature (PCT) for the design basis LOCA at power uprate conditions. In order to support these future elements of the Full Potential Program, analyses for containment conversion have been performed to incorporate inputs and assumptions that support power uprate to 2910 MWT and steam generator replacement on BVPS-1. In this way, analyses performed to support containment conversion such as mass and energy release analysis and containment pressure and temperature response analysis also apply to operation of BVPS-1 and BVPS-2 at the uprated power level of 2910 MWT and to BVPS-1 with replacement steam generators. This report presents the results of the work performed to support conversion of the containment pressure design basis from sub-atmospheric to atmospheric, including analytical inputs and assumptions that also support the power uprate and steam generator replacement elements of the Full Potential Program. The remaining work required to support implementation of power uprate and steam generator replacement will be addressed in separate reports and submittals to be provided in the future.

Containment analyses to support containment conversion utilized the MAAP5 program along with Westinghouse mass and energy release data. The containment methodology included in the MAAP5 program has recently been submitted to the NRC for approval in Westinghouse WCAP-15844, Revision 0. The analyses show the calculated pressures and temperatures are

within the current structural design. Thus, there was no need for structural analysis, nor was there a reduction in the structural analysis margins. Likewise, the containment subcompartment analysis shows that the pressures are within the current design and thus there was no need for structural analysis nor was 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) in accordance with 10CFR50.67, Standard Review Plan 15.0.1, and Regulatory Guide 1.183.

The containment analyses supporting conversion to an atmospheric containment were done at the future uprated NSSS power level (2910 MWT). However, the units will operate with atmospheric containments at the current NSSS power level (2697 MWT) until separate power uprate licensing amendments are approved by the NRC.

Certain aspects of this application, when granted and implemented, will allow increased operational flexibility and efficiency, a reduction in regulatory requirements, and a reduction in calculated radiological doses for specific design basis accidents.

This report addresses conversion to an atmospheric containment and selective use of the AST. Power uprate for BVPS-1 and BVPS-2 will be addressed in separate submittals to the NRC.

1.1 Proposed Changes

Proposed methodology changes discussed in this report include the following:

- Change 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.
- Change the Main Steamline Break (MSLB) mass and energy release calculation methodology to that documented in WCAP-8822.
- Change the Containment Integrity and NPSH (BVPS-1 only) calculation methodology from LOCTIC to MAAP5.
- Implement a "selective" application of the Alternative Source Term (AST) in accordance with RG 1.183 for dose analyses impacted by containment conversion.

Changes in parameters currently controlled by Technical Specifications include:

- Increase in the containment operating pressure range to 12 psia to 16 psia total pressure and eliminate the Maximum Allowable Operating (MAO) Air Partial Pressure Curve from the Technical Specifications.

- Decrease the RWST level setpoints for transfer from the injection to the recirculation mode.
- Increase the Containment High Pressure setpoint (CIA).
- Increase the Containment Intermediate High-High Pressure setpoint.
- Increase the Containment High-High Pressure setpoint (CIB).
- Change the calculated LOCA accident peak containment pressure (P_a) to reflect the analysis results.
- Decrease the minimum containment temperature operating limit.
- Elimination of the Quench Spray flow reduction (cutback) (BVPS-1 only).
- Reduction in River/Service water flow requirements to the recirculation spray heat exchangers.
- Increase in the minimum flow requirement for the hydrogen recombiners (BVPS-2).
- Revised acceptance criteria for the BVPS-1 control room HEPA and charcoal filters which will make the BVPS acceptance criteria similar to the more limiting criteria currently listed for BVPS-2.

Changes eliminating system operability requirements in the Technical Specifications include:

- Removal of the Control Room Emergency Bottled Air Pressurization System (CREBAPS) operability requirements from the Technical Specifications.

Other 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.

1.2 Benefits of the Changes

Conversion to an atmospheric containment will provide margin in the Large Break LOCA calculated PCT to support uprating the NSSS power to 2910 MWT.

The changes will make containment entry safer by eliminating the need for air packs which allows for more frequent containment entry and inspections. This change should significantly reduce worker fatigue and temperature stress and increase mobility/agility and visibility as well

as enhance communications. Thus, the time required to perform inspections is reduced (ALARA) and inspections will also be more effective.

The change will eliminate the station startup delay associated with establishing containment vacuum prior to entry into Mode 4. The change will also eliminate the need for vacuum pump discharges associated with maintaining the containment sub-atmospheric.

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

The changes avail the plant of the most recent technology improvements/methods (MAAP5 and Westinghouse mass and energy releases, and use of the Alternative Source Term (AST)). The containment methodology utilized in the MAAP5 program has been submitted for NRC review and approval. All other methodologies utilized in this program have been approved for other plants in previous submittals.

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 Westinghouse mass and energy releases and the MAAP5 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.

- Evaluation of the post accident radiological consequences of atmospheric conversion using the “selective” implementation of Alternative Source Term in accordance with 10CFR50.67 and Regulatory Guide 1.183.

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

Chapter 2.0 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 MAAP5 and Westinghouse mass and energy release methods are provided in Chapter 3.0 and Chapter 4.0. Chapter 5.0 presents a discussion of the supporting radiological analyses. Chapters 6.0 and 7.0 discuss the evaluations and analyses performed to

investigate the effects of the proposed changes on other safety analyses, systems, and programs. Chapter 8.0 discusses a review of the environmental effects of the proposed changes.

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.

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 BVPS-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.

The radiological analysis for these accidents, which is currently based on Total Integrated Dose (TID) 14844 and NUREG 0800 methodology, demonstrates compliance with the dose limits of 10CFR100.11 at the site boundary (EAB and LPZ) and GDC 19 for Control Room Habitability.

2.0 NSSS PARAMETERS

2.1 Nuclear Steam Supply System (NSSS) Parameters

The containment conversion project and power uprate project 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 in general the analyses performed for containment conversion apply to power uprate 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 power uprate conditions with the Model 54F RSGs.

2.1.1 NSSS Design (PCWG) Parameters

2.1.1.1 Introduction

The NSSS PCWG parameters are the fundamental parameters which 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 design transients, systems, components, accidents, and fuel analyses and evaluations.

The NSSS design 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 therefore established for the vessel average temperature (Tavg) (i.e., RCS average temperature), feedwater temperature, and the SGTP level. The Tavg 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 uprated parameters. These uprated 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 future power uprate conditions.

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

- The power level for the uprating was set at 2910 MWT NSSS (2900 MWT reactor).
- The future uprated 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 fuel with IFMs.
- Both 0% and 22% SGTP is assumed.
- Design core bypass flow is 6.5%.
- Parameters support the existing steam generators for BVPS-1 and BVPS-2 and replacement Model 54F steam generators for BVPS-1.
- A range of full power normal operating Tavg from 566.2°F to 580.0°F is selected for the analyses. This allows the plant to operate anywhere within this range.
- A range of full power feedwater temperature of 400°F to 455°F is selected for the analyses.

2.1.1.3 Description of Analyses and Evaluations

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

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

Table 2.1-2 lists the parameters for BVPS-1 and BVPS-2 and reflects the current steam generators. These parameters incorporate the future uprated NSSS power level of 2910 MWT, 0% and 22% SGTP, Tavg 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 generator. These parameters incorporate the future uprated NSSS power level of 2910 MWT, 0% and 22% SGTP, Tavg 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 future power uprate 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 design 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.

2.1.1.5 Conclusions

The PCWG parameters for the containment conversion project incorporate power uprate conditions which bound the current operating conditions and the future power uprating.

Table 2.1-1
Design Power Capability Parameters
BVPS-1 and BVPS-2 - Current Parameters

THERMAL DESIGN PARAMETERS	**Case 1**	**Case 2**
NSSS Power % MWT 10^6 BTU/hr	101.4 2697 9203	101.4 2697 9203
Reactor Power MWT 10^6 BTU/hr	2689 9175	2689 9175
Thermal Design Flow, loop gpm Reactor 10^6 lb/hr	87,200 99.5	87,200 99.5
Reactor Coolant Pressure, psia	2250	2250
Core Bypass, %	6.5	6.5
Reactor Coolant Temperature, °F Core Outlet Vessel Outlet Core Average Vessel Average Vessel/Core Inlet Steam Generator Outlet	 615.1 610.8 580.3 576.2 541.6 541.3	 615.1 610.8 580.3 576.2 541.6 541.3
Steam Generator Steam Temperature, °F Steam Pressure, psia Steam Flow, 10^6 lb/hr total Feed Temperature, °F Moisture, % max. Tube Plugging Level (%)	 519.0 806 11.81 439.3 0.25 0	 505.5(1) 716(1) 11.78 439.3 0.25 30
Zero Load Temperature, °F	547	547
HYDRAULIC DESIGN PARAMETERS		
Pump Design Point, Flow (gpm)/head (ft)	88,500/280	
Mechanical Design Flow, gpm	101,400	
Minimum Measured Flow, gpm/total	266,800	

Notes:

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

Table 2.1-2
Design Power Capability Parameters
BVPS-1 and BVPS-2 - Power Uprating

	Power Uprating			
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	493.3	518.1	508.5
Steam Pressure, psia	699	641	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, gpm	101,400			
Minimum Measured Flow, gpm/total	266,800			

Table 2.1-3
Design Power Capability Parameters
BVPS-1 - Power Uprating/Model 54F RSGs

THERMAL DESIGN PARAMETERS	Power Uprating/Model 54F RSGs			
	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	500.5	490.2	515.8	505.6
Steam Pressure, psia	684	623	783	716
Steam Flow, 10^6 lb/hr total	12.04/12.99	12.02/12.97	12.08/13.03	12.05/13.00
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, gpm	101,400			
Minimum Measured Flow, gpm/total	266,800			

3.0 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 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 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 51 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 Units.

The Westinghouse generated LOCA mass and energy releases for the first hour were used in the MAAP5 containment response analysis. After this time, the break enthalpy is calculated, along with the containment response, by MAAP5.

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 is +15 percent. 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 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, which is conservatively chosen to

bound both steam generator types. 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) and an allowance for calorimetric error (+0.6 percent of power) is included.

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.

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.

Thus, based on the previously discussed conditions and assumptions, a separate but bounding analysis for each of the Beaver Valley 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 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.

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, Section 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 BVPS-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 ANS Condition IV event, an infrequent fault. To satisfy the NRC acceptance criteria presented in the Standard Review Plan 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 (Reference 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 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 MAAP5 code. The MAAP5 code was used for convenience. This method of determining the long-term mass and energy releases is consistent with past applications of Westinghouse methodology. These user defined functions are characterized for 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 that are used are those calculated by the EPITOME 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 containment cools. The user defined functions for the outlet specific enthalpies are as follows:

Unmixed: $h_{out,1} = (Q_{decay} + Q_{sg,1}) / W_1 + h_{in}$ [3.1-1]

$h_{out,2} = (Q_{sg,23}) / W_2 + h_{in}$ [3.1-2]

Mixed: $h_{out} = (Q_{decay} + Q_{sg,1} + Q_{sg,23}) / (W_1 + W_2) + h_{in}$ [3.1-3]

In these expressions, Q_{decay} is the ANS 1979 core decay heat (Table 3.1-12, Part 1), $Q_{sg,1}$ is the release rate of stored energy for the steam generator in the affected loop (from the upper head internals and upper elliptical heads), $Q_{sg,23}$ is the release rate of remaining energy from the generators in the intact loops, W_1 is the steam flow rate through the affected generator calculated by the DBA LOCA code, W_2 is the total RCS injection flow rate minus W_1, and h_{in} is the recirculation water specific enthalpy entering the RCS.

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 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

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 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.

The MAAP5 Containment Evaluation Model (Chapter 4.0) was used to calculate the sump temperature. The results of the sump temperature response calculation are presented in Section 4.1.7.7.

3.1.7 Conclusions

BVPS-1 and BVPS-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.

3.1.8 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. 7106), for D. C. Cook Nuclear Plant BVPS-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.

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)	833.47
Steam Generator Design	54F
Steam Generator Tube Plugging (percent)	0
Initial Steam Generator Secondary Side Mass (lbm)*	111035
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

Note:

A maximum Safety Injection Temperature of 65°F was used during the Injection Phase and 190°F was used during the Recirculation Phase.

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

Note:

A maximum Safety Injection Temperature of 65°F was used during the Injection Phase and 210°F was conservatively used during the Recirculation Phase.

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)	(Btu/sec)	(lbm/sec)	(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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
00000	.0	.0	.0	.0
.00112	84619.5	45648.7	40274.9	21681.8
.101	40363.3	21812.1	20325.9	10935.1
.202	47387.5	25815.1	22494.7	12110.4
.302	47225.6	25990.2	23019.3	12404.9
.402	47739.9	26595.2	22700.7	12247.2
.502	46937.3	26503.5	21998.4	11878.2
.601	44642.6	25541.6	21329.0	11523.3
.702	45159.8	26151.2	20838.2	11262.1
.802	44898.4	26274.5	20464.3	11063.8
.901	43918.1	25948.1	20177.9	10913.0
1.00	42703.6	25466.5	19966.9	10801.9
1.10	41539.9	24999.1	19806.1	10717.1
1.20	40443.5	24555.7	19713.7	10668.5
1.30	39414.7	24137.2	19694.7	10659.3
1.40	38524.9	23780.4	19720.2	10673.6
1.50	37781.3	23493.6	19716.7	10671.4
1.60	37135.1	23251.9	19679.6	10650.3
1.70	36498.6	23013.8	19634.8	10624.9
1.80	35823.8	22758.9	19604.6	10607.5
1.90	35100.4	22491.0	19555.1	10579.9

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
2.00	34301.4	22194.8	19451.3	10522.8
2.10	33243.5	21746.8	19286.9	10432.9
2.20	32054.6	21219.6	19061.7	10310.1
2.30	30757.6	20621.6	18720.3	10125.1
2.40	29385.1	19963.4	18400.6	9952.0
2.50	27245.6	18737.3	18054.0	9764.4
2.60	23620.0	16419.3	17782.7	9618.5
2.70	21430.9	15078.3	17510.5	9472.6
2.80	20100.5	14265.7	17202.7	9307.6
2.90	18708.7	13335.5	16914.7	9154.0
3.00	17662.3	12632.7	16668.9	9023.9
3.10	16815.3	12065.0	16451.2	8909.4
3.20	16076.8	11568.1	16236.6	8796.7
3.30	15429.9	11134.7	16017.6	8682.0
3.40	14876.0	10768.9	15870.0	8606.0
3.50	14357.7	10427.6	15680.5	8507.2
3.60	13884.5	10117.9	15503.9	8415.4
3.70	13464.3	9844.8	15344.8	8333.3
3.80	13097.0	9605.8	15188.2	8252.5
3.90	12784.2	9400.8	15046.2	8179.7
4.00	12510.4	9217.7	14898.9	8103.8
4.20	12042.3	8897.3	14642.8	7973.3
4.40	11664.8	8637.4	14396.2	7847.8

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
4.60	11329.9	8406.3	14167.6	7732.1
4.80	11051.0	8216.1	13916.2	7604.0
5.00	10816.2	8056.9	13451.0	7357.7
5.20	10618.9	7928.1	14423.7	7908.8
5.40	10404.0	7791.2	14781.0	8107.1
5.60	10171.4	7647.3	14541.3	7985.9
5.80	10008.5	7542.2	14367.3	7899.5
6.00	10336.1	7786.6	14165.1	7797.9
6.20	10473.7	8115.8	13872.6	7645.2
6.40	9192.5	7894.3	13501.2	7448.1
6.60	8239.2	7493.6	13338.5	7365.2
6.80	7998.7	7282.3	13122.1	7249.2
7.00	7896.0	7117.0	12797.4	7068.9
7.20	7904.0	6980.6	12525.8	6916.0
7.40	7979.8	6860.2	12417.3	6853.4
7.60	8049.0	6749.6	12305.5	6786.4
7.80	8076.4	6662.6	12143.1	6687.9
8.00	8057.2	6554.9	11941.3	6567.1
8.20	8054.5	6456.3	11761.4	6461.1
8.40	8046.1	6359.0	11613.2	6374.7
8.60	8012.6	6259.1	11448.9	6279.5
8.80	7953.2	6159.2	11265.8	6174.2
9.00	7869.4	6061.7	11106.9	6083.1

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
9.20	7759.9	5964.2	10956.1	5997.2
9.40	7628.7	5867.9	10790.8	5903.7
9.60	7480.1	5771.8	10629.3	5812.9
9.80	7327.4	5684.5	10479.3	5729.2
10.0	7161.5	5595.0	10309.1	5635.0
10.2	6993.9	5504.0	10141.8	5543.1
10.4	6829.1	5413.4	9981.5	5455.6
10.6	6666.1	5322.7	9816.5	5365.7
10.8	6506.8	5232.7	9653.7	5277.1
11.0	6339.9	5134.5	9480.6	5182.8
11.2	6167.0	5029.6	9292.9	5080.7
11.4	5994.4	4918.0	9123.3	4989.7
11.6	5832.2	4800.9	8954.5	4898.8
11.8	5685.3	4685.7	8773.9	4801.0
12.0	5548.1	4572.5	8608.1	4711.5
12.2	5409.7	4457.2	8427.4	4613.9
12.4	5271.8	4343.3	8257.7	4522.7
12.6	5133.4	4229.0	8083.9	4429.4
12.8	4999.6	4119.0	7768.1	4256.6
13.0	4882.0	4022.3	7582.9	4159.1
13.2	4774.9	3934.9	7357.3	4034.0
13.4	4675.4	3860.2	7195.0	3920.8
13.6	4570.4	3789.0	7065.1	3802.0

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
13.8	4466.7	3730.3	6994.0	3698.0
14.0	4360.5	3678.7	7091.4	3675.2
14.2	4240.7	3629.1	6814.3	3457.1
14.4	4115.3	3587.9	6680.4	3312.4
14.6	3975.8	3546.7	6440.4	3127.0
14.8	3826.0	3511.4	6050.6	2878.3
15.0	3669.1	3481.1	5849.1	2726.4
15.2	3500.3	3453.0	5658.0	2592.6
15.4	3311.8	3421.1	5428.9	2450.7
15.6	3019.6	3303.8	5087.5	2263.5
15.8	2677.6	3130.2	4631.1	2026.1
16.0	2404.6	2926.7	4383.4	1878.9
16.2	2162.3	2663.3	4130.2	1736.0
16.4	1966.9	2434.7	3867.7	1597.0
16.6	1808.3	2245.4	3546.0	1439.0
16.8	1661.2	2067.5	3091.0	1232.6
17.0	1533.1	1912.1	2694.8	1052.0
17.2	1422.2	1776.9	2376.7	903.7
17.4	1307.7	1636.2	2173.5	804.2
17.6	1199.2	1503.0	2055.5	740.4
17.8	1096.2	1375.7	1983.9	696.0
18.0	999.0	1255.2	1978.8	676.1
18.2	788.8	992.0	1981.8	658.6

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
18.4	546.6	688.4	1705.2	556.2
18.6	381.5	481.4	1406.3	454.6
18.8	250.3	316.3	1091.3	351.6
19.0	160.3	202.9	880.1	283.7
19.2	78.8	99.9	748.4	241.7
19.4	.0	.0	798.7	258.5
19.6	.0	.0	876.1	282.9
19.8	.0	.0	912.1	292.6
20.0	.0	.0	909.7	290.2
20.2	.0	.0	853.7	271.6
20.4	.0	.0	750.8	238.9
20.6	.0	.0	600.9	191.7
20.8	.0	.0	354.8	113.8
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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
21.0	.0	.0	.0	.0
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	64.8	76.3	.0	.0
22.1	33.0	38.9	.0	.0
22.2	34.0	40.1	.0	.0
22.3	38.2	45.0	.0	.0
22.4	47.8	56.3	.0	.0
22.6	55.8	65.7	.0	.0
22.7	62.8	73.9	.0	.0
22.8	69.6	81.9	.0	.0
22.9	76.3	89.9	.0	.0
23.0	81.2	95.7	.0	.0
23.1	85.9	101.2	.0	.0
23.2	90.4	106.5	.0	.0
23.3	94.7	111.6	.0	.0
23.4	98.9	116.5	.0	.0
23.5	102.9	121.3	.0	.0
23.6	106.8	125.9	.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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
23.7	110.6	130.3	.0	.0
23.8	114.3	134.7	.0	.0
23.9	117.9	138.9	.0	.0
24.0	121.4	143.0	.0	.0
24.1	124.8	147.0	.0	.0
25.1	154.9	182.6	.0	.0
26.1	454.3	538.0	4314.6	492.1
26.7	456.1	540.2	4314.4	498.7
27.1	479.5	567.9	4638.3	487.4
28.1	469.3	556.0	4459.0	497.8
29.1	459.4	544.2	4366.4	489.7
30.1	449.5	532.4	4273.0	481.4
31.0	440.8	522.0	4190.3	473.9
31.1	439.9	520.9	4181.4	473.1
32.1	430.7	509.9	4092.5	465.0
33.1	421.9	499.4	4006.8	457.2
34.1	413.5	489.3	3924.4	449.6
35.1	405.4	479.8	3845.1	442.3
36.1	397.8	470.7	3768.9	435.3
36.8	392.7	464.6	3717.3	430.6
37.1	390.5	462.0	3695.7	428.6
38.1	383.6	453.7	3625.2	422.0
39.1	376.9	445.8	3557.3	415.8

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
40.1	370.6	438.2	3491.9	409.7
41.1	364.5	431.0	3428.7	403.9
42.1	358.6	424.0	3367.7	398.2
43.1	353.0	417.3	3308.8	392.7
43.6	350.3	414.1	3279.9	390.1
44.1	347.6	410.9	3251.7	387.4
45.1	342.4	404.7	3196.5	382.3
46.1	337.4	398.8	3142.9	377.3
47.1	332.6	393.0	3091.0	372.5
48.1	327.9	387.5	3040.5	367.7
49.1	323.4	382.1	2991.5	363.2
50.1	319.0	376.9	2943.8	358.7
51.1	314.8	371.9	2897.4	354.3
51.2	314.4	371.4	2892.7	353.9
52.1	310.7	367.0	2852.2	350.1
53.1	306.7	362.3	2808.1	345.9
54.1	302.8	357.7	2765.1	341.9
55.1	299.1	353.3	2723.2	337.9
56.2	238.7	281.7	1981.1	275.5
57.2	276.1	325.9	240.1	119.8
58.2	273.7	323.1	239.1	118.6
59.2	268.1	316.5	237.2	116.0
60.2	262.7	310.1	235.3	113.4

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
61.2	257.3	303.7	233.5	110.9
62.2	252.0	297.4	231.6	108.4
63.2	246.7	291.1	229.8	106.0
64.2	241.9	285.4	228.2	103.8
65.2	237.2	279.9	226.6	101.7
66.2	232.7	274.5	225.1	99.6
67.2	228.3	269.3	223.6	97.6
68.2	224.0	264.2	222.2	95.7
69.2	219.8	259.2	220.8	93.9
70.2	215.7	254.5	219.4	92.1
71.2	211.8	249.8	218.1	90.4
72.2	208.0	245.3	216.8	88.7
73.2	204.2	240.9	215.6	87.1
74.2	200.6	236.6	214.4	85.6
75.2	197.1	232.4	213.2	84.1
76.2	193.7	228.4	212.1	82.6
77.2	190.4	224.5	211.1	81.2
78.2	187.2	220.7	210.0	79.9
79.2	184.0	217.0	209.0	78.6
80.2	181.0	213.4	208.1	77.4
81.2	178.1	210.0	207.1	76.2
82.2	175.3	206.6	206.2	75.0
84.2	169.9	200.2	204.5	72.8

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
86.2	164.8	194.3	203.0	70.8
88.2	160.2	188.8	201.6	69.0
90.2	155.9	183.7	200.3	67.3
92.2	151.9	179.0	199.1	65.8
92.5	151.3	178.3	198.9	65.6
94.2	148.2	174.7	198.0	64.4
96.2	144.9	170.7	197.0	63.1
98.2	141.8	167.1	196.1	62.0
100.2	139.0	163.8	195.3	60.9
102.2	136.5	160.8	194.5	60.0
104.2	134.1	158.1	193.8	59.1
106.2	132.1	155.6	193.2	58.4
108.2	130.2	153.4	192.7	57.7
110.2	128.5	151.4	192.2	57.1
112.2	127.0	149.7	191.8	56.5
114.2	125.7	148.1	191.4	56.0
116.2	124.5	146.7	191.1	55.6
118.2	123.5	145.5	190.8	55.2
120.2	122.6	144.5	190.5	54.9
121.2	122.2	144.0	190.4	54.7
122.2	121.9	143.6	190.3	54.6
124.2	121.2	142.8	190.1	54.3
126.2	120.6	142.1	189.9	54.1

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
128.2	120.2	141.6	189.7	53.9
130.2	119.8	141.1	189.6	53.7
132.2	119.5	140.7	189.5	53.6
134.2	119.2	140.5	189.4	53.5
136.2	119.0	140.2	189.3	53.4
138.2	118.9	140.1	189.3	53.3
140.2	118.8	140.0	189.2	53.2
142.2	118.8	139.9	189.2	53.2
144.2	118.8	140.0	189.1	53.1
146.2	118.8	140.0	189.1	53.1
148.2	118.9	140.1	189.1	53.1
150.2	119.0	140.2	189.1	53.1
152.2	119.1	140.3	189.1	53.1
153.0	119.2	140.4	189.1	53.1
154.2	119.2	140.5	189.1	53.1
156.2	119.4	140.7	189.1	53.1
158.2	119.6	140.9	189.2	53.2
160.2	119.8	141.1	189.2	53.2
162.2	120.0	141.3	189.2	53.2
164.2	120.2	141.6	189.2	53.3
166.2	120.4	141.8	189.3	53.3
168.2	120.6	142.1	189.3	53.3
170.2	120.9	142.4	189.3	53.4

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
172.2	121.1	142.7	189.4	53.4
174.2	121.4	143.0	189.4	53.5
176.2	122.0	143.7	189.6	53.7
178.2	122.7	144.5	190.1	54.0
180.2	123.2	145.1	191.0	54.3
182.2	123.6	145.6	192.2	54.6
184.2	124.0	146.1	193.7	55.1
186.2	124.3	146.4	195.4	55.5
186.5	124.3	146.5	195.7	55.6
188.2	124.5	146.7	197.4	56.0
190.2	124.6	146.8	199.5	56.5
192.2	124.7	146.9	201.9	57.0
194.2	124.6	146.8	204.4	57.6
196.2	124.4	146.6	207.0	58.1
198.2	124.1	146.3	209.8	58.7
200.2	123.7	145.8	212.7	59.2
202.2	123.2	145.1	215.8	59.8
204.2	122.5	144.3	219.0	60.4
206.2	121.7	143.4	222.3	61.0
208.2	120.8	142.3	225.8	61.6
210.2	119.7	141.0	229.4	62.2
212.2	118.5	139.6	233.2	62.8
214.2	117.1	138.0	237.0	63.4

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

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
216.2	115.6	136.2	241.0	64.0
218.2	113.9	134.2	245.1	64.6
220.2	112.1	132.0	249.3	65.3
222.2	111.7	131.6	251.6	65.4

* 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	217.2	.000	.000	.00	.00	.333	.0	.0	.0	.00
21.7	213.3	22.915	.000	.54	1.95	.000	7093.2	7093.2	.0	74.49
21.9	209.9	27.967	.000	1.07	1.90	.000	6998.2	6998.2	.0	74.49
23.1	207.8	2.906	.315	1.50	6.18	.450	6632.9	6632.9	.0	74.49
23.9	207.3	2.816	.430	1.62	9.41	.469	6425.8	6425.8	.0	74.49
26.1	205.9	5.076	.618	1.92	15.62	.694	5411.5	5411.5	.0	74.49
26.7	205.3	4.858	.648	2.01	15.63	.692	5281.2	5281.2	.0	74.49
27.1	205.0	4.924	.663	2.07	15.63	.732	5595.7	5211.5	.0	71.66
31.0	202.7	4.222	.725	2.50	15.63	.692	4977.9	4590.7	.0	71.28
36.8	201.3	3.736	.747	3.00	15.63	.676	4392.0	3991.8	.0	70.73
43.6	201.2	3.385	.754	3.50	15.63	.660	3877.7	3466.9	.0	70.12
51.2	202.3	3.107	.756	4.01	15.63	.643	3431.2	3012.0	.0	69.45
56.2	203.3	2.620	.753	4.30	15.63	.582	2407.1	1972.0	.0	67.03
57.2	203.6	2.856	.756	4.36	15.51	.616	427.5	.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)
60.2	204.6	2.730	.755	4.53	14.97	.612	429.4	.0	.0	33.22
69.2	209.1	2.365	.751	5.00	13.75	.594	435.7	.0	.0	33.22
80.2	216.5	2.038	.747	5.51	12.88	.571	439.6	.0	.0	33.22
92.5	225.5	1.789	.745	6.00	12.48	.546	442.2	.0	.0	33.22
108.2	235.2	1.611	.744	6.57	12.52	.522	443.8	.0	.0	33.22
121.2	241.9	1.540	.746	7.00	12.81	.512	444.4	.0	.0	33.22
138.2	249.4	1.502	.750	7.54	13.33	.508	444.6	.0	.0	33.22
153.0	255.0	1.494	.755	8.00	13.84	.509	444.7	.0	.0	33.22
158.2	256.8	1.494	.756	8.16	14.02	.509	444.7	.0	.0	33.22
170.2	260.8	1.495	.760	8.52	14.44	.512	444.6	.0	.0	33.22
186.5	265.6	1.509	.766	9.00	14.99	.518	444.4	.0	.0	33.22
204.2	270.1	1.478	.771	9.51	15.39	.517	444.6	.0	.0	33.22
222.2	274.1	1.382	.774	10.00	15.58	.502	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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
222.3	126.7	158.8	318.9	75.8
227.3	126.3	158.4	319.2	75.8
232.3	125.9	157.9	319.6	75.7
237.3	125.5	157.4	320.0	75.7
242.3	126.1	158.2	319.4	75.3
247.3	125.8	157.7	319.8	75.3
252.3	125.4	157.3	320.1	75.2
257.3	126.0	158.0	319.6	74.9
262.3	125.6	157.5	319.9	74.9
267.3	125.2	157.0	320.3	74.8
272.3	125.8	157.7	319.7	74.5
277.3	125.4	157.3	320.1	74.4
282.3	125.0	156.8	320.5	74.4
287.3	124.6	156.3	320.9	74.3
292.3	125.2	157.0	320.3	74.0
297.3	124.8	156.5	320.7	74.0
302.3	124.4	156.0	321.1	73.9
307.3	124.9	156.7	320.6	73.6
312.3	124.5	156.2	321.0	73.5
317.3	124.1	155.7	321.4	73.5
322.3	124.7	156.3	320.9	73.2
327.3	124.3	155.8	321.3	73.1
332.3	123.9	155.3	321.7	73.1

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
337.3	124.4	156.0	321.2	72.8
342.3	124.0	155.5	321.6	72.7
347.3	124.5	156.1	321.1	72.4
352.3	124.0	155.6	321.5	72.4
357.3	123.6	155.0	321.9	72.3
362.3	124.1	155.6	321.4	72.0
367.3	123.7	155.1	321.9	72.0
372.3	123.3	154.6	322.3	71.9
377.3	123.7	155.1	321.8	71.6
382.3	123.3	154.6	322.3	71.6
387.3	122.8	154.1	322.7	71.5
392.3	123.3	154.6	322.2	71.2
397.3	122.9	154.1	322.7	71.2
402.3	123.3	154.7	322.2	70.9
407.3	123.0	154.3	322.5	70.8
412.3	122.7	153.9	322.8	70.7
417.3	123.2	154.5	322.3	70.4
422.3	122.9	154.1	322.6	70.3
427.3	122.6	153.7	323.0	70.3
432.3	123.1	154.3	322.5	69.9
437.3	122.7	153.9	322.8	69.9
442.3	122.4	153.5	323.1	69.8
447.3	122.9	154.1	322.6	69.5

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
452.3	122.5	153.7	323.0	69.4
457.3	122.2	153.2	323.4	69.4
462.3	122.7	153.8	322.9	69.1
467.3	122.3	153.4	323.2	69.0
472.3	121.9	152.9	323.6	68.9
477.3	122.4	153.5	323.2	68.6
482.3	122.0	153.0	323.5	68.6
487.3	122.4	153.5	323.1	68.3
492.3	122.1	153.1	323.5	68.2
497.3	121.7	152.6	323.9	68.1
502.3	122.1	153.1	323.5	67.8
507.3	121.7	152.6	323.9	67.8
512.3	122.1	153.1	323.5	67.5
517.3	121.7	152.6	323.9	67.4
522.3	122.0	153.0	323.5	67.2
527.3	121.6	152.5	323.9	69.9
532.3	121.2	152.0	324.3	69.8
537.3	121.5	152.4	324.0	69.5
542.3	121.1	151.9	324.4	69.4
547.3	121.4	152.3	324.1	69.2
552.3	121.0	151.7	324.6	69.1
557.3	121.3	152.1	324.3	68.8
562.3	121.6	152.5	324.0	68.6

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
567.3	121.1	151.9	324.4	68.5
572.3	121.4	152.2	324.2	68.2
577.3	120.9	151.6	324.7	68.2
582.3	121.1	151.9	324.4	67.9
587.3	120.6	151.2	324.9	67.9
592.3	120.8	151.5	324.7	67.6
597.3	121.0	151.7	324.5	67.4
602.3	120.5	151.1	325.0	67.3
607.3	120.7	151.4	324.8	67.1
612.3	120.9	151.6	324.6	66.8
617.3	120.4	150.9	325.2	66.8
622.3	120.5	151.1	325.0	66.5
627.3	120.7	151.3	324.9	66.3
632.3	120.8	151.4	324.8	66.1
637.3	120.2	150.7	325.3	66.0
642.3	120.3	150.8	325.3	65.8
647.3	120.3	150.9	325.2	65.6
652.3	120.3	150.9	325.2	68.0
657.3	120.3	150.9	325.2	67.8
662.3	120.3	150.9	325.2	67.6
667.3	120.3	150.8	325.3	67.4
672.3	120.2	150.7	325.4	67.2
677.3	120.1	150.6	325.5	67.0

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
682.3	119.9	150.4	325.6	66.8
687.3	119.7	150.2	325.8	66.7
692.3	120.1	150.6	325.4	66.4
697.3	119.9	150.3	325.7	66.2
702.3	120.1	150.6	325.4	65.9
707.3	119.8	150.2	325.8	65.8
712.3	119.9	150.4	325.6	65.5
717.3	120.0	150.5	325.5	65.3
722.3	119.5	149.8	326.1	65.2
727.3	119.9	150.4	325.6	64.9
732.3	119.7	150.2	325.8	64.7
737.3	119.5	149.8	326.1	64.5
742.3	119.6	150.0	325.9	66.8
747.3	119.6	149.9	326.0	66.5
752.3	119.4	149.7	326.2	66.3
757.3	119.5	149.8	326.1	66.1
762.3	119.3	149.6	326.2	65.9
767.3	119.3	149.6	326.2	65.6
772.3	119.3	149.6	326.2	65.4
777.3	119.2	149.5	326.3	65.2
782.3	119.2	149.5	326.3	64.9
787.3	67.5	84.6	378.1	78.2
1244.8	67.5	84.6	378.1	78.2

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

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
1244.9	64.6	80.5	381.0	73.9
1247.3	64.5	80.5	381.0	73.8
1526.2	64.5	80.5	381.0	73.8
1526.3	57.2	65.8	388.3	12.9
2900.0	49.5	57.0	396.0	13.2
2900.1	51.7	59.5	356.2	60.3
3600.0	48.2	55.5	359.7	60.9

* 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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
21.0	.0	.0	.0	.0
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	64.8	76.3	.0	.0
22.1	33.0	38.9	.0	.0
22.2	34.0	40.1	.0	.0
22.3	38.2	45.0	.0	.0
22.4	47.8	56.3	.0	.0
22.6	55.8	65.7	.0	.0
22.7	62.8	73.9	.0	.0
22.8	69.6	81.9	.0	.0
22.9	76.3	89.9	.0	.0
23.0	81.2	95.7	.0	.0
23.1	85.9	101.2	.0	.0
23.2	90.4	106.5	.0	.0
23.3	94.7	111.6	.0	.0
23.4	98.9	116.5	.0	.0
23.5	102.9	121.3	.0	.0
23.6	106.8	125.9	.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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
23.7	110.6	130.3	.0	.0
23.8	114.3	134.7	.0	.0
23.9	117.9	138.9	.0	.0
24.0	121.4	143.0	.0	.0
24.1	124.8	147.0	.0	.0
25.1	154.9	182.6	.0	.0
26.1	454.3	538.0	4314.6	492.1
26.7	456.1	540.2	4314.4	498.7
27.1	500.1	592.4	4889.0	478.8
28.1	487.7	578.1	4644.3	503.6
29.1	477.9	566.3	4553.7	495.7
30.1	468.1	554.5	4461.7	487.4
30.9	460.4	545.3	4389.2	480.8
31.1	458.5	543.1	4371.3	479.2
32.1	449.2	532.0	4283.5	471.2
33.1	440.4	521.5	4198.8	463.4
34.1	431.9	511.4	4117.3	455.9
35.1	423.9	501.8	4038.8	448.7
36.1	416.2	492.6	3963.4	441.7
36.5	413.2	489.1	3934.1	439.0
37.1	408.9	483.9	3890.9	435.0
38.1	401.9	475.6	3821.1	428.6
39.1	395.2	467.6	3753.8	422.3

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
40.1	388.8	460.0	3689.0	416.3
41.1	382.7	452.6	3626.5	410.5
42.1	376.8	445.6	3566.1	404.9
43.0	371.7	439.5	3513.3	400.0
43.1	371.1	438.9	3507.7	399.5
44.1	365.7	432.4	3451.1	394.2
45.1	360.4	426.2	3396.4	389.1
46.1	355.4	420.1	3343.4	384.2
47.1	350.5	414.3	3291.9	379.4
48.1	345.8	408.7	3241.9	374.7
49.1	341.2	403.3	3193.4	370.1
50.1	336.8	398.1	3146.1	365.7
50.3	335.9	397.0	3136.8	364.8
51.1	332.5	393.0	3100.2	361.4
52.1	328.4	388.0	3055.4	357.2
53.1	324.3	383.2	3011.8	353.0
54.1	320.4	378.6	2969.3	349.0
55.1	316.6	374.1	2927.8	345.1
56.1	255.8	301.9	2210.8	284.1
57.1	219.8	259.3	330.0	98.7
58.1	164.5	193.9	420.6	96.2
59.1	164.2	193.5	421.2	96.0
60.1	163.8	193.1	421.9	95.9

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
61.1	163.5	192.7	422.6	95.7
62.1	163.2	192.3	423.3	95.5
63.1	162.9	192.0	424.0	95.4
64.1	162.5	191.6	424.7	95.2
65.1	162.2	191.2	425.4	95.1
66.1	161.9	190.8	426.1	94.9
67.1	161.5	190.4	426.8	94.7
68.1	161.2	190.0	427.5	94.6
69.1	160.9	189.6	428.2	94.4
70.0	160.6	189.3	428.9	94.3
70.1	160.6	189.3	428.9	94.3
71.1	160.3	188.9	429.7	94.1
72.1	159.9	188.5	430.4	94.0
73.1	159.6	188.1	431.1	93.8
74.1	159.3	187.8	431.8	93.7
75.1	159.0	187.4	432.5	93.5
76.1	158.7	187.0	433.2	93.4
77.1	158.4	186.6	434.0	93.2
78.1	158.0	186.3	434.7	93.1
79.1	157.7	185.9	435.4	92.9
80.1	157.4	185.5	436.1	92.8
81.1	157.1	185.2	436.8	92.6
82.1	156.8	184.8	437.6	92.5

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
84.1	156.1	184.0	439.0	92.2
86.1	155.5	183.3	440.5	91.9
88.1	154.9	182.5	442.0	91.7
90.1	154.2	181.8	443.5	91.4
92.1	153.6	181.0	445.0	91.1
93.7	153.1	180.4	446.2	90.9
94.1	153.0	180.3	446.5	90.9
96.1	152.3	179.5	448.0	90.6
98.1	151.7	178.8	449.5	90.3
100.1	151.0	178.0	451.0	90.1
102.1	150.4	177.2	452.5	89.8
104.1	149.7	176.5	454.0	89.5
106.1	149.1	175.7	455.5	89.2
108.1	148.4	174.9	456.9	89.0
110.1	147.8	174.1	458.4	88.7
112.1	147.1	173.3	459.9	88.4
114.1	146.4	172.5	461.4	88.1
116.1	145.7	171.8	462.8	87.9
118.1	145.1	171.0	464.3	87.6
119.5	144.6	170.4	465.3	87.4
120.1	144.4	170.2	465.8	87.3
122.1	143.7	169.4	467.2	87.0
124.1	143.0	168.5	468.7	86.8

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
126.1	142.3	167.7	470.2	86.5
128.1	141.7	166.9	471.6	86.2
130.1	141.0	166.1	473.1	85.9
132.1	140.3	165.3	474.5	85.7
134.1	139.6	164.5	476.0	85.4
136.1	138.9	163.7	477.4	85.1
138.1	138.2	162.8	478.9	84.8
140.1	137.5	162.0	480.3	84.6
142.1	136.8	161.2	481.7	84.3
144.1	136.1	160.4	483.2	84.0
146.1	135.4	159.5	484.6	83.7
147.9	134.8	158.8	485.9	83.5
148.1	134.7	158.7	486.0	83.5
150.1	134.0	157.9	487.4	83.2
152.1	133.3	157.0	488.9	82.9
154.1	132.5	156.2	490.3	82.6
156.1	131.8	155.3	491.7	82.4
158.1	131.1	154.5	493.2	82.1
160.1	130.4	153.6	494.6	81.8
162.1	129.7	152.8	496.0	81.6
164.1	128.9	151.9	497.4	81.3
166.1	128.2	151.1	498.9	81.0
168.1	127.5	150.2	500.3	80.8

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**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
170.1	126.8	149.4	501.7	80.5
172.1	126.0	148.5	503.1	80.2
174.1	125.3	147.6	504.5	80.0
176.1	124.6	146.8	505.9	79.7
178.1	123.8	145.9	507.3	79.4
179.6	123.3	145.3	508.4	79.2
180.1	123.1	145.0	508.7	79.2
182.1	122.4	144.2	510.1	78.9
184.1	121.9	143.6	511.0	78.9
186.1	121.4	143.0	512.0	78.9
188.1	120.9	142.4	512.9	78.9
190.1	120.4	141.8	513.8	78.9
192.1	119.9	141.2	514.7	78.9
194.1	119.4	140.6	515.7	78.9
196.1	118.9	140.0	516.6	78.8
198.1	118.4	139.4	517.5	78.8
200.1	117.9	138.9	518.4	78.8
202.1	117.4	138.3	519.3	78.8
204.1	116.9	137.7	520.2	78.7
206.1	116.4	137.1	521.1	78.7
208.1	115.9	136.5	522.0	78.7
210.1	115.4	135.9	522.9	78.6
212.1	114.9	135.4	523.8	78.6

Table 3.1-9 (Continued)
Double-Ended Pump Suction Break Maximum Safeguards 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)
214.1	114.4	134.8	524.7	78.5
215.3	114.1	134.5	525.3	78.5

* 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)

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	217.2	.000	.000	.00	.00	.333	.0	.0	.0	.00
21.7	213.3	22.915	.000	.54	1.95	.000	7093.2	7093.2	.0	74.49
21.9	209.9	27.967	.000	1.07	1.90	.000	6998.2	6998.2	.0	74.49
23.1	207.8	2.906	.315	1.50	6.18	.450	6632.9	6632.9	.0	74.49
23.9	207.3	2.816	.430	1.62	9.41	.469	6425.8	6425.8	.0	74.49
26.1	205.9	5.076	.618	1.92	15.62	.694	5411.5	5411.5	.0	74.49
26.7	205.3	4.858	.648	2.01	15.63	.692	5281.2	5281.2	.0	74.49
27.1	205.0	5.068	.663	2.07	15.63	.764	5869.3	5211.5	.0	69.86
30.9	202.6	4.362	.725	2.51	15.63	.699	5208.1	4552.8	.0	69.30
36.5	201.0	3.879	.747	3.00	15.63	.685	4640.4	3972.4	.0	68.55
43.0	200.7	3.532	.755	3.50	15.63	.670	4143.4	3464.9	.0	67.73
50.3	201.5	3.254	.757	4.00	15.63	.655	3707.6	3020.7	.0	66.84
57.1	202.8	2.359	.753	4.43	15.63	.535	707.4	.0	.0	33.22
59.1	203.4	2.120	.744	4.52	15.63	.501	717.3	.0	.0	33.22
70.0	208.2	2.066	.746	5.00	15.63	.502	717.3	.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)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
82.1	215.8	2.007	.748	5.52	15.63	.503	717.3	.0	.0	33.22
93.7	224.4	1.949	.750	6.00	15.63	.504	717.3	.0	.0	33.22
108.1	234.2	1.878	.754	6.57	15.63	.505	717.3	.0	.0	33.22
119.5	240.8	1.821	.756	7.00	15.63	.505	717.3	.0	.0	33.22
134.1	248.1	1.749	.759	7.53	15.63	.506	717.3	.0	.0	33.22
147.9	254.0	1.682	.761	8.00	15.63	.506	717.3	.0	.0	33.22
164.1	259.9	1.603	.764	8.53	15.63	.506	717.3	.0	.0	33.22
179.6	264.8	1.529	.767	9.00	15.63	.506	717.4	.0	.0	33.22
198.1	269.6	1.456	.770	9.53	15.63	.508	717.4	.0	.0	33.22
215.3	273.5	1.392	.773	10.00	15.63	.510	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)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
215.4	129.9	162.9	588.2	85.5
220.4	129.5	162.4	588.6	85.5
225.4	130.1	163.1	588.0	85.1
230.4	129.7	162.6	588.4	85.1
235.4	129.3	162.2	588.8	85.0
240.4	128.9	161.7	589.2	85.0
245.4	129.5	162.4	588.6	84.6
250.4	129.1	161.9	589.0	84.6
255.4	128.7	161.4	589.4	84.5
260.4	129.3	162.1	588.8	84.2
265.4	128.9	161.6	589.2	84.1
270.4	128.5	161.1	589.6	84.1
275.4	129.0	161.8	589.1	83.8
280.4	128.6	161.3	589.5	83.7
285.4	128.2	160.7	589.9	83.7
290.4	128.7	161.4	589.4	83.3
295.4	128.3	160.9	589.8	83.3
300.4	128.8	161.5	589.3	83.0
305.4	128.4	161.0	589.7	82.9
310.4	128.0	160.5	590.1	82.9
315.4	128.5	161.1	589.6	82.6
320.4	128.1	160.6	590.0	82.5
325.4	127.6	160.0	590.5	82.4

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

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
330.4	128.1	160.6	590.0	82.1
335.4	127.7	160.1	590.4	82.1
340.4	127.2	159.5	590.9	82.0
345.4	127.7	160.1	590.4	81.7
350.4	127.3	159.6	590.8	81.7
355.4	127.7	160.1	590.4	81.4
360.4	127.3	159.6	590.8	81.3
365.4	126.8	159.0	591.3	81.3
370.4	127.2	159.6	590.9	81.0
375.4	126.8	159.0	591.3	80.9
380.4	127.2	159.5	590.9	80.6
385.4	126.7	158.9	591.4	80.6
390.4	127.1	159.4	591.0	80.3
395.4	126.7	158.8	591.4	80.3
400.4	126.2	158.2	591.9	80.2
405.4	126.7	158.9	591.4	79.9
410.4	126.3	158.4	591.8	79.8
415.4	126.8	159.0	591.3	79.5
420.4	126.4	158.6	591.7	79.4
425.4	126.1	158.1	592.0	79.4
430.4	126.5	158.7	591.6	79.1
435.4	126.2	158.2	591.9	79.0
440.4	125.8	157.7	592.3	78.9

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

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
445.4	126.2	158.3	591.9	78.6
450.4	125.8	157.8	592.3	78.6
455.4	126.2	158.3	591.9	78.3
460.4	125.8	157.8	592.3	78.2
465.4	126.2	158.3	591.9	77.9
470.4	125.8	157.8	592.3	77.8
475.4	125.4	157.2	592.7	77.8
480.4	125.8	157.7	592.3	77.5
485.4	125.3	157.2	592.8	77.4
490.4	125.7	157.6	592.4	77.2
495.4	125.3	157.1	592.8	77.1
500.4	125.6	157.5	592.5	76.8
505.4	125.1	156.9	593.0	76.8
510.4	125.5	157.3	592.6	76.5
515.4	125.0	156.7	593.1	76.4
520.4	125.3	157.1	592.8	76.2
525.4	125.5	157.4	592.6	78.6
530.4	125.0	156.8	593.1	78.6
535.4	125.3	157.1	592.8	78.3
540.4	124.8	156.5	593.3	78.2
545.4	125.0	156.7	593.1	78.0
550.4	125.2	157.0	592.9	77.7
555.4	124.7	156.3	593.4	77.7

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

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
560.4	124.8	156.5	593.3	77.4
565.4	125.0	156.7	593.1	77.2
570.4	124.4	156.0	593.7	77.1
575.4	124.5	156.2	593.6	76.9
580.4	124.6	156.3	593.5	76.7
585.4	124.7	156.4	593.4	76.4
590.4	124.8	156.5	593.3	76.2
595.4	124.1	155.7	594.0	76.2
600.4	124.2	155.7	593.9	76.0
605.4	124.2	155.7	593.9	75.8
610.4	124.2	155.8	593.9	75.6
615.4	124.2	155.8	593.9	75.4
620.4	124.2	155.7	593.9	75.2
625.4	124.1	155.6	594.0	75.0
630.4	124.0	155.5	594.1	74.8
635.4	123.9	155.3	594.2	74.6
640.4	123.7	155.1	594.4	77.0
645.4	124.1	155.7	594.0	76.7
650.4	123.9	155.3	594.2	76.5
655.4	123.6	155.0	594.5	76.4
660.4	123.9	155.3	594.2	76.1
665.4	124.1	155.6	594.0	75.8
670.4	123.6	155.0	594.5	75.7

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

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
675.4	123.7	155.1	594.4	75.4
680.4	123.7	155.1	594.4	75.2
685.4	123.6	155.0	594.5	75.0
690.4	123.4	154.8	594.7	74.8
695.4	123.7	155.1	594.4	74.5
700.4	123.3	154.6	594.8	74.4
705.4	123.2	154.5	594.9	74.2
710.4	123.5	154.9	594.6	73.9
715.4	123.1	154.4	595.0	73.8
720.4	123.4	154.8	594.7	73.4
725.4	123.0	154.2	595.1	75.8
730.4	123.1	154.3	595.0	75.5
735.4	123.1	154.3	595.0	75.2
740.4	122.9	154.1	595.2	75.0
745.4	122.8	154.0	595.3	74.8
750.4	68.3	85.6	649.8	88.8
1203.5	68.3	85.6	649.8	88.8
1203.6	64.5	80.5	653.6	83.7
1205.4	64.5	80.5	653.6	83.7
1513.8	64.5	80.5	653.6	83.7
1513.9	56.0	64.4	662.1	22.0
2500.0	49.9	57.4	668.2	22.2
2500.1	53.1	61.1	613.0	113.4

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)
3600.0	47.6	54.8	618.5	114.4

* 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

Table 3.1-12 (Part 1) (Continued) LOCA Mass and Energy Release Analysis ANS 1979 Core Decay Heat Power Fraction	
Time (sec)	**ANS 1979 Decay Heat Fraction**
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)	222.23(3)	1244.92(4)	1526.16(5)	3600.00(6)
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	621.13	621.13	621.13	621.13	621.13	621.13	621.13
Added Mass	Pumped Injection	.00	.00	.00	85.34	540.95	666.25	1563.88
	Total Added	.00	.00	.00	85.34	540.95	666.25	1563.88
Total Available		621.13	621.13	621.13	706.47	1162.08	1287.38	2185.01
Distribution	Reactor Coolant	420.62	42.07	56.84	102.51	102.51	102.51	102.51
	Accumulator	200.51	160.35	145.57	.00	.00	.00	.00
	Total Contents	621.13	202.41	202.41	102.51	102.51	102.51	102.51
Effluent	Break Flow	.00	418.70	418.70	595.27	1050.88	1176.18	2073.81
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	418.70	418.70	595.27	1050.88	1176.18	2073.81
Total Accountable		621.13	621.12	621.12	697.77	1153.39	1278.69	2176.32

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)	215.33(3)	1203.63(4)	1513.79(5)	3600.00(6)
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	621.13	621.13	621.13	621.13	621.13	621.13	621.13
Added Mass	Pumped Injection	.00	.00	.00	133.79	843.44	1066.17	2507.07
	Total Added	.00	.00	.00	133.79	843.44	1066.17	2507.07
Total Available		621.13	621.13	621.13	754.92	1464.57	1687.30	3128.20
Distribution	Reactor Coolant	420.62	42.07	56.84	102.71	102.71	102.71	102.71
	Accumulator	200.51	160.35	145.57	.00	.00	.00	.00
	Total Contents	621.13	202.41	202.41	102.71	102.71	102.71	102.71
Effluent	Break Flow	.00	418.70	418.70	643.52	1353.17	1575.89	3016.80
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	418.70	418.70	643.52	1353.17	1575.89	3016.80
Total Accountable		621.13	621.12	621.12	746.23	1455.88	1678.60	3119.51

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)	222.23(3)	1244.92(4)	1526.16(5)	3600.00(6)
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	632.36	632.36	632.36	632.36	632.36	632.36	632.36
Added Energy	Pumped Injection	.00	.00	.00	2.83	17.97	22.13	87.62
	Decay Heat	.00	5.73	5.73	25.49	94.76	110.65	209.75
	Heat From Secondary	.00	.38	.38	.38	4.10	4.14	4.14
	Total Added	.00	6.12	6.12	28.71	116.83	136.92	301.51
Total Available		632.36	638.47	638.47	661.06	749.18	769.28	933.87
Distribution	Reactor Coolant	247.64	10.39	11.49	26.81	26.81	26.81	26.81
	Accumulator	14.95	11.95	10.85	.00	.00	.00	.00
	Core Stored	22.89	12.51	12.51	3.91	3.20	3.14	2.71
	Primary Metal	124.37	117.25	117.25	95.72	52.45	46.94	40.72
	Secondary Metal	39.94	40.33	40.33	36.60	20.83	17.82	15.51
	Steam Generator	182.56	184.83	184.83	164.22	91.00	77.95	68.09
	Total Contents	632.36	377.26	377.26	327.28	194.29	172.66	153.84
Effluent	Break Flow	.00	260.72	260.72	325.46	546.57	569.03	754.00
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	260.72	260.72	325.46	546.57	569.03	754.00
Total Accountable		632.36	637.99	637.99	652.74	740.86	741.69	907.83

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)	215.33(3)	1203.63(4)	1513.79(5)	3600.00(6)
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	632.36	632.36	632.36	632.36	632.36	632.36	632.36
Added Energy	Pumped Injection	.00	.00	.00	4.44	28.02	35.42	189.55
	Decay Heat	.00	5.73	5.73	24.90	92.34	109.96	209.71
	Heat From Secondary	.00	.38	.38	.38	3.97	4.04	4.04
	Total Added	.00	6.12	6.12	29.73	124.33	149.42	403.30
Total Available		632.36	638.47	638.47	662.09	756.69	781.77	1035.66
Distribution	Reactor Coolant	247.64	10.39	11.49	26.84	26.84	26.84	26.84
	Accumulator	14.95	11.95	10.85	.00	.00	.00	.00
	Core Stored	22.89	12.51	12.51	3.91	3.13	3.08	2.71
	Primary Metal	124.37	117.25	117.25	95.21	51.95	45.90	40.66
	Secondary Metal	39.94	40.33	40.33	36.45	20.74	17.41	15.47
	Steam Generator	182.56	184.83	184.83	163.34	90.51	76.07	67.82
	Total Contents	632.36	377.26	377.26	325.74	193.18	169.29	153.49
Effluent	Break Flow	.00	260.72	260.72	328.03	555.20	584.51	855.40
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	260.72	260.72	328.03	555.20	584.51	855.40
Total Accountable		632.36	637.99	637.99	653.77	748.37	753.80	1008.89

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

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
Sump Temperature 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)	(Btu/sec)	(lbm/sec)	(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

Table 3.1-19 (Continued)
Sump Temperature 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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

Table 3.1-19 (Continued)
Sump Temperature 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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

Table 3.1-19 (Continued)
Sump Temperature 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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

Table 3.1-19 (Continued)
Sump Temperature 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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

Table 3.1-19 (Continued)
Sump Temperature 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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
Sump Temperature 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*		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	.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
Sump Temperature 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

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)
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
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

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)
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
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	6226.6

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 and 150°F was used for the SW failure case.

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)	(Btu/sec)	(lbm/sec)	(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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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)

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	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

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)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
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
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

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)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
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
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

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)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
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
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

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)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
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
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

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)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
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

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)
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
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

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)
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
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

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)
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
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

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)
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
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

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)
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

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)
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

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)
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
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*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
214.2	115.2	145.0	353.1	77.7
219.2	115.9	145.9	352.5	77.4
224.2	115.5	145.5	352.8	77.3
229.2	115.2	145.1	353.1	77.3
234.2	114.9	144.7	353.4	77.2
239.2	115.6	145.5	352.8	76.9
244.2	115.2	145.1	353.1	76.8
249.2	114.9	144.7	353.4	76.8
254.2	114.6	144.3	353.7	76.7
259.2	115.2	145.1	353.1	76.4
264.2	114.9	144.7	353.4	76.3
269.2	114.6	144.3	353.7	76.3
274.2	114.3	143.9	354.1	76.2
279.2	114.9	144.7	353.4	75.9
284.2	114.6	144.3	353.7	75.8
289.2	114.2	143.9	354.1	75.8
294.2	113.9	143.4	354.4	75.7
299.2	114.5	144.2	353.8	75.4
304.2	114.2	143.8	354.1	75.3
309.2	113.9	143.4	354.4	75.3
314.2	114.5	144.2	353.8	75.0
319.2	114.2	143.8	354.1	74.9

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)
324.2	113.8	143.3	354.5	74.8
329.2	113.5	142.9	354.8	74.8
334.2	114.1	143.7	354.2	74.5
339.2	113.8	143.2	354.6	74.4
344.2	113.4	142.8	354.9	74.3
349.2	113.1	142.4	355.2	74.3
354.2	113.7	143.1	354.6	74.0
359.2	113.3	142.7	355.0	73.9
364.2	113.0	142.3	355.3	73.8
369.2	113.6	143.0	354.7	73.5
374.2	113.2	142.6	355.1	73.5
379.2	112.9	142.1	355.5	73.4
384.2	112.5	141.7	355.8	73.3
389.2	113.1	142.4	355.2	73.0
394.2	112.7	141.9	355.6	73.0
399.2	112.4	141.5	355.9	72.9
404.2	113.0	142.3	355.3	72.6
409.2	112.8	142.0	355.5	72.5
414.2	112.6	141.7	355.8	72.4
419.2	112.3	141.4	356.0	72.3
424.2	112.1	141.1	356.2	72.2
429.2	112.8	142.0	355.6	74.6
434.2	112.5	141.7	355.8	74.5

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)
439.2	112.3	141.4	356.1	74.4
444.2	112.0	141.0	356.3	74.3
449.2	111.8	140.7	356.5	74.2
454.2	112.4	141.6	355.9	73.9
459.2	112.2	141.2	356.1	73.8
464.2	111.9	140.9	356.4	73.7
469.2	111.7	140.6	356.6	73.6
474.2	112.3	141.4	356.0	73.3
479.2	112.0	141.1	356.3	73.2
484.2	111.8	140.8	356.5	73.1
489.2	111.5	140.4	356.8	73.0
494.2	112.2	141.2	356.2	72.7
499.2	111.9	140.9	356.4	72.6
504.2	111.6	140.6	356.7	72.5
509.2	111.4	140.2	357.0	72.4
514.2	111.1	139.9	357.2	72.3
519.2	111.7	140.6	356.6	71.9
524.2	111.4	140.3	356.9	71.8
529.2	111.1	140.0	357.2	71.7
534.2	111.7	140.7	356.6	71.4
539.2	111.5	140.3	356.9	71.3
544.2	111.2	140.0	357.1	71.2
549.2	110.9	139.6	357.4	71.1

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)
554.2	111.5	140.4	356.8	70.8
559.2	111.2	140.0	357.1	70.7
564.2	110.9	139.6	357.4	70.6
569.2	110.6	139.3	357.7	70.5
574.2	111.2	140.0	357.2	70.2
579.2	110.9	139.6	357.4	70.1
584.2	110.6	139.2	357.7	70.0
589.2	111.1	139.9	357.2	69.7
594.2	110.8	139.5	357.5	69.6
599.2	110.5	139.1	357.8	69.5
604.2	110.2	138.8	358.1	71.9
609.2	110.8	139.5	357.5	71.6
614.2	110.5	139.2	357.8	71.5
619.2	110.2	138.8	358.1	71.4
624.2	110.8	139.5	357.5	71.0
629.2	110.5	139.1	357.8	70.9
634.2	110.2	138.8	358.1	70.8
639.2	110.7	139.5	357.6	70.5
644.2	110.5	139.1	357.9	70.4
649.2	110.2	138.7	358.1	70.2
654.2	109.9	138.4	358.4	70.1
659.2	110.4	139.0	357.9	69.8
664.2	110.1	138.6	358.2	69.7

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)
669.2	109.8	138.2	358.5	69.6
674.2	110.3	138.8	358.1	69.3
679.2	109.9	138.4	358.4	69.1
684.2	110.4	139.0	357.9	68.8
689.2	110.1	138.6	358.2	68.7
694.2	109.8	138.2	358.5	68.6
699.2	110.2	138.8	358.1	68.3
704.2	109.9	138.3	358.4	68.2
709.2	109.5	137.9	358.8	70.5
714.2	109.9	138.4	358.4	70.2
719.2	109.6	138.0	358.7	70.1
724.2	110.0	138.5	358.3	69.8
729.2	109.6	138.1	358.7	69.6
734.2	109.3	137.6	359.0	69.5
739.2	109.7	138.1	358.7	69.2
744.2	109.3	137.6	359.0	69.1
749.2	109.6	138.0	358.7	68.8
754.2	109.2	137.6	359.1	68.7
759.2	109.6	138.0	358.7	68.4
764.2	109.2	137.5	359.1	68.2
769.2	109.5	137.8	358.8	68.0
774.2	109.1	137.3	359.2	67.8
779.2	109.3	137.7	359.0	67.5

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)
784.2	108.9	137.1	359.4	67.4
789.2	109.2	137.5	359.1	67.2
794.2	109.4	137.7	358.9	69.2
799.2	108.9	137.2	359.4	69.1
804.2	109.2	137.5	359.1	68.8
809.2	109.4	137.7	358.9	68.5
814.2	108.9	137.1	359.4	68.4
819.2	109.1	137.4	359.2	68.1
824.2	109.2	137.6	359.1	67.8
829.2	108.7	136.9	359.6	67.7
834.2	108.9	137.1	359.4	67.4
839.2	109.0	137.2	359.3	67.2
844.2	109.0	137.3	359.3	66.9
849.2	108.5	136.6	359.8	66.8
854.2	108.5	136.7	359.8	66.5
859.2	108.5	136.7	359.8	68.5
864.2	108.5	136.7	359.8	68.3
869.2	108.5	136.6	359.8	68.0
874.2	108.4	136.5	359.9	67.8
879.2	108.3	136.4	360.0	67.5
884.2	108.7	136.9	359.6	67.2
889.2	108.5	136.7	359.8	66.9
894.2	108.3	136.4	360.0	66.7

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)
899.2	65.4	82.4	402.9	77.7
1180.4	65.4	82.4	402.9	77.7
1180.5	65.1	81.5	403.2	75.0
1184.2	65.0	81.5	403.3	75.8
1494.7	65.0	81.5	403.3	75.8
1494.8	56.6	65.1	411.8	13.6
2948.0	48.4	55.7	419.9	13.9
2948.1	50.9	58.6	453.5	40.0
3600.0	47.9	55.1	456.5	40.2

* 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

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)
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
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

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)
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
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

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)
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
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

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)
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
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

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)
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
176.5	113.2	133.4	506.4	79.8
178.5	112.5	132.6	507.8	79.5

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)
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 (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	4.389	.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

Table 3.1-37 (Continued)
Principle Parameters During Reflood Double-Ended Pump Suction Break - Maximum Safeguards
(BVPS-2) (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)
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
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	117.7	148.3	595.6	87.3
218.4	117.4	147.9	595.9	87.2
223.4	117.1	147.5	596.3	87.1
228.4	117.8	148.4	595.6	86.8
233.4	117.4	148.0	595.9	86.7
238.4	117.1	147.6	596.2	86.7
243.4	116.8	147.1	596.6	86.6
248.4	117.4	148.0	595.9	86.3
253.4	117.1	147.6	596.2	86.2
258.4	116.8	147.2	596.6	86.1
263.4	116.4	146.7	596.9	86.1
268.4	117.1	147.6	596.2	85.7
273.4	116.8	147.2	596.6	85.7
278.4	116.4	146.7	596.9	85.6
283.4	117.1	147.6	596.2	85.3
288.4	116.7	147.1	596.6	85.2
293.4	116.4	146.7	596.9	85.1
298.4	116.1	146.3	597.3	85.1
303.4	116.7	147.1	596.6	84.8
308.4	116.4	146.6	597.0	84.7
313.4	116.0	146.2	597.3	84.6
318.4	115.7	145.8	597.7	84.5

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)
323.4	116.3	146.6	597.0	84.2
328.4	116.0	146.1	597.4	84.2
333.4	115.6	145.7	597.7	84.1
338.4	115.3	145.2	598.1	84.0
343.4	115.9	146.0	597.5	83.7
348.4	115.5	145.6	597.8	83.6
353.4	115.2	145.1	598.2	83.6
358.4	115.8	145.9	597.6	83.2
363.4	115.4	145.4	597.9	83.2
368.4	115.1	145.0	598.3	83.1
373.4	114.7	144.5	598.6	83.0
378.4	115.3	145.3	598.1	82.7
383.4	114.9	144.8	598.4	82.7
388.4	114.6	144.4	598.8	82.6
393.4	115.1	145.1	598.2	82.3
398.4	114.8	144.6	598.6	82.2
403.4	114.5	144.3	598.9	82.1
408.4	114.2	144.0	599.1	82.0
413.4	114.9	144.8	598.4	81.7
418.4	114.7	144.5	598.7	81.6
423.4	114.4	144.2	598.9	81.5
428.4	114.2	143.9	599.2	81.4
433.4	114.8	144.7	598.5	81.1

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)
438.4	114.6	144.4	598.8	81.0
443.4	114.3	144.1	599.0	80.9
448.4	114.1	143.7	599.3	80.8
453.4	113.8	143.4	599.5	80.7
458.4	114.5	144.2	598.9	80.4
463.4	114.2	143.9	599.1	80.3
468.4	113.9	143.6	599.4	80.2
473.4	113.7	143.2	599.7	80.1
478.4	114.3	144.0	599.0	79.7
483.4	114.0	143.7	599.3	79.6
488.4	113.7	143.4	599.6	79.6
493.4	113.5	143.0	599.9	79.5
498.4	114.1	143.8	599.2	79.1
503.4	113.8	143.4	599.5	79.0
508.4	113.5	143.1	599.8	78.9
513.4	113.3	142.7	600.1	81.5
518.4	113.9	143.5	599.5	81.1
523.4	113.6	143.1	599.8	81.0
528.4	113.3	142.8	600.1	80.9
533.4	113.0	142.4	600.3	80.8
538.4	113.6	143.1	599.8	80.5
543.4	113.3	142.8	600.1	80.4
548.4	113.0	142.4	600.4	80.3

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)
553.4	113.5	143.1	599.8	79.9
558.4	113.2	142.7	600.1	79.8
563.4	112.9	142.3	600.4	79.7
568.4	113.5	143.0	599.9	79.4
573.4	113.2	142.6	600.2	79.3
578.4	112.9	142.2	600.5	79.2
583.4	112.5	141.8	600.8	79.1
588.4	113.1	142.5	600.3	78.8
593.4	112.8	142.1	600.6	78.7
598.4	112.4	141.7	600.9	78.6
603.4	113.0	142.4	600.4	78.3
608.4	112.7	142.0	600.7	78.1
613.4	112.4	141.6	600.9	78.0
618.4	112.9	142.3	600.4	77.7
623.4	112.6	141.9	600.7	77.6
628.4	112.3	141.6	601.0	77.5
633.4	112.8	142.2	600.5	77.2
638.4	112.5	141.8	600.8	77.1
643.4	112.2	141.4	601.1	76.9
648.4	112.7	142.0	600.6	79.1
653.4	112.4	141.6	600.9	79.0
658.4	112.1	141.2	601.3	78.9
663.4	112.5	141.8	600.8	78.5

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)
668.4	112.2	141.4	601.1	78.4
673.4	111.9	141.0	601.5	78.3
678.4	112.3	141.5	601.0	78.0
683.4	112.0	141.1	601.4	77.9
688.4	112.4	141.6	600.9	77.5
693.4	112.0	141.2	601.3	77.4
698.4	111.7	140.8	601.7	77.3
703.4	112.1	141.2	601.3	77.0
708.4	111.7	140.8	601.6	76.9
713.4	112.1	141.3	601.2	76.6
718.4	111.7	140.8	601.6	76.5
723.4	112.1	141.2	601.3	76.2
728.4	111.7	140.7	601.7	76.1
733.4	112.0	141.1	601.3	75.8
738.4	111.6	140.6	601.7	78.0
743.4	111.9	141.0	601.4	77.7
748.4	111.5	140.5	601.9	77.6
753.4	111.8	140.8	601.6	77.3
758.4	111.3	140.3	602.0	77.2
763.4	111.6	140.6	601.8	76.9
768.4	111.1	140.1	602.2	76.8
773.4	111.4	140.3	602.0	76.5
778.4	111.6	140.6	601.8	76.2

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)
783.4	111.1	140.0	602.3	76.1
788.4	111.3	140.2	602.1	75.8
793.4	111.4	140.4	601.9	75.5
798.4	110.9	139.8	602.4	75.4
803.4	111.0	139.9	602.3	75.1
808.4	111.2	140.1	602.2	74.9
813.4	111.3	140.2	602.1	76.9
818.4	111.3	140.3	602.0	76.6
823.4	110.8	139.6	602.6	76.5
828.4	110.8	139.7	602.5	76.2
833.4	110.8	139.7	602.5	76.0
838.4	110.8	139.7	602.5	75.7
843.4	110.8	139.6	602.6	75.5
848.4	110.7	139.5	602.6	75.2
853.4	110.6	139.4	602.7	75.0
858.4	111.0	139.9	602.3	74.6
863.4	110.8	139.7	602.5	74.4
868.4	110.6	139.4	602.7	76.4
873.4	110.9	139.7	602.5	76.1
878.4	65.8	82.9	647.5	87.6
1157.6	65.8	82.9	647.5	87.6
1157.7	64.8	81.3	648.6	84.3
1158.4	64.8	81.3	648.6	84.2

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)
1517.0	64.8	81.3	648.6	84.2
1517.1	54.5	62.8	658.8	21.7
2782.0	47.3	54.4	666.0	22.0
2782.1	49.7	57.2	454.7	40.1
3600.0	46.0	52.9	458.4	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)	1180.47(4)	1494.67(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	85.24	537.75	684.90	1694.38
	Total Added	.00	.00	.00	85.24	537.75	684.90	1694.38
Total Available		626.47	626.47	626.47	711.71	1164.23	1311.37	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	1038.04	1185.19	2194.67
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	417.78	417.78	585.53	1038.04	1185.19	2194.67
Total Accountable		626.47	626.46	626.46	702.86	1155.38	1302.52	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)	1157.73(4)	1517.04(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	804.32	1060.62	2375.56
	Total Added	.00	.00	.00	130.69	804.32	1060.62	2375.56
Total Available		626.47	626.47	626.47	757.16	1430.79	1687.09	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	1304.41	1560.71	2875.67
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	417.78	417.78	630.78	1304.41	1560.71	2875.67
Total Accountable		626.47	626.46	626.46	748.31	1421.94	1678.24	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)	1180.47(4)	1494.67(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	17.75	22.60	74.05
	Decay Heat	.00	5.79	5.79	24.82	90.99	108.91	209.67
	Heat From Secondary	.00	.43	.43	.43	10.95	11.03	11.03
	Total Added	.00	6.22	6.22	28.06	119.68	142.54	294.76
Total Available		640.99	647.21	647.21	669.04	760.67	783.52	935.74
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	3.11	2.71
	Primary Metal	115.85	109.61	109.61	88.92	50.92	45.49	40.39
	Secondary Metal	34.40	34.82	34.82	31.86	19.28	16.51	14.79
	Steam Generator	207.00	210.21	210.21	188.90	118.56	102.62	92.91
	Total Contents	640.99	388.24	388.24	342.76	221.10	196.90	179.97
Effluent	Break Flow	.00	258.48	258.48	318.23	531.52	557.66	728.55
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	258.48	258.48	318.23	531.52	557.66	728.55
Total Accountable		640.99	646.72	646.72	660.99	752.61	754.56	908.51

Table 3.1-43 (Continued)
Double-Ended Pump Suction Break, Energy Balance Minimum Safeguards
(BVPS-2) (Case 1L)

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)	1157.73(4)	1517.04(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	26.54	35.00	101.15
	Decay Heat	.00	5.79	5.79	24.75	89.64	110.13	209.64
	Heat From Secondary	.00	.43	.43	.43	10.71	10.86	10.86
	Total Added	.00	6.22	6.22	29.49	126.89	155.99	321.65
Total Available		640.99	647.21	647.21	670.47	767.88	796.98	962.64
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	3.04	2.71
	Primary Metal	115.85	109.61	109.61	88.61	50.43	44.25	40.35
	Secondary Metal	34.40	34.82	34.82	31.84	19.20	16.02	14.76
	Steam Generator	207.00	210.21	210.21	188.76	117.87	99.63	92.59
	Total Contents	640.99	388.24	388.24	342.31	219.78	192.12	179.58

Table 3.1-44 (Continued)
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)	1157.73(4)	1517.04(5)	3600.00(6)
		Energy (Million Btu)						
Effluent	Break Flow	.00	258.48	258.48	320.12	540.05	575.52	755.26
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	258.48	258.48	320.12	540.05	575.52	755.26
Total Accountable		640.99	646.72	646.72	662.42	759.83	767.64	934.85

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)	(Btu/sec)	(lbm/sec)	(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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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
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

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)	(Btu/sec)	(lbm/sec)	(Btu/sec)
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 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	4.389	.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

Table 3.1-46 (Continued)
Double-Ended Pump Suction (SW Failure) - Maximum Safeguards Principle Parameters During Reflood (BVPS-2) (Case 2L1)

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)
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
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)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
213.4	117.7	148.3	595.6	87.3
218.4	117.4	147.9	595.9	87.2
223.4	117.1	147.5	596.3	87.1
228.4	117.8	148.4	595.6	86.8
233.4	117.4	148.0	595.9	86.7
238.4	117.1	147.6	596.2	86.7
243.4	116.8	147.1	596.6	86.6
248.4	117.4	148.0	595.9	86.3
253.4	117.1	147.6	596.2	86.2
258.4	116.8	147.2	596.6	86.1
263.4	116.4	146.7	596.9	86.1
268.4	117.1	147.6	596.2	85.7
273.4	116.8	147.2	596.6	85.7
278.4	116.4	146.7	596.9	85.6
283.4	117.1	147.6	596.2	85.3
288.4	116.7	147.1	596.6	85.2
293.4	116.4	146.7	596.9	85.1
298.4	116.1	146.3	597.3	85.1
303.4	116.7	147.1	596.6	84.8
308.4	116.4	146.6	597.0	84.7
313.4	116.0	146.2	597.3	84.6
318.4	115.7	145.8	597.7	84.5

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)
323.4	116.3	146.6	597.0	84.2
328.4	116.0	146.1	597.4	84.2
333.4	115.6	145.7	597.7	84.1
338.4	115.3	145.2	598.1	84.0
343.4	115.9	146.0	597.5	83.7
348.4	115.5	145.6	597.8	83.6
353.4	115.2	145.1	598.2	83.6
358.4	115.8	145.9	597.6	83.2
363.4	115.4	145.4	597.9	83.2
368.4	115.1	145.0	598.3	83.1
373.4	114.7	144.5	598.6	83.0
378.4	115.3	145.3	598.1	82.7
383.4	114.9	144.8	598.4	82.7
388.4	114.6	144.4	598.8	82.6
393.4	115.1	145.1	598.2	82.3
398.4	114.8	144.6	598.6	82.2
403.4	114.5	144.3	598.9	82.1
408.4	114.2	144.0	599.1	82.0
413.4	114.9	144.8	598.4	81.7
418.4	114.7	144.5	598.7	81.6
423.4	114.4	144.2	598.9	81.5
428.4	114.2	143.9	599.2	81.4
433.4	114.8	144.7	598.5	81.1

Table 3.1-47 (Continued)
Double-Ended Pump Suction (SW Failure) Maximum Safeguards
Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
438.4	114.6	144.4	598.8	81.0
443.4	114.3	144.1	599.0	80.9
448.4	114.1	143.7	599.3	80.8
453.4	113.8	143.4	599.5	80.7
458.4	114.5	144.2	598.9	80.4
463.4	114.2	143.9	599.1	80.3
468.4	113.9	143.6	599.4	80.2
473.4	113.7	143.2	599.7	80.1
478.4	114.3	144.0	599.0	79.7
483.4	114.0	143.7	599.3	79.6
488.4	113.7	143.4	599.6	79.6
493.4	113.5	143.0	599.9	79.5
498.4	114.1	143.8	599.2	79.1
503.4	113.8	143.4	599.5	79.0
508.4	113.5	143.1	599.8	78.9
513.4	113.3	142.7	600.1	81.5
518.4	113.9	143.5	599.5	81.1
523.4	113.6	143.1	599.8	81.0
528.4	113.3	142.8	600.1	80.9
533.4	113.0	142.4	600.3	80.8
538.4	113.6	143.1	599.8	80.5
543.4	113.3	142.8	600.1	80.4
548.4	113.0	142.4	600.4	80.3

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)
553.4	113.5	143.1	599.8	79.9
558.4	113.2	142.7	600.1	79.8
563.4	112.9	142.3	600.4	79.7
568.4	113.5	143.0	599.9	79.4
573.4	113.2	142.6	600.2	79.3
578.4	112.9	142.2	600.5	79.2
583.4	112.5	141.8	600.8	79.1
588.4	113.1	142.5	600.3	78.8
593.4	112.8	142.1	600.6	78.7
598.4	112.4	141.7	600.9	78.6
603.4	113.0	142.4	600.4	78.3
608.4	112.7	142.0	600.7	78.1
613.4	112.4	141.6	600.9	78.0
618.4	112.9	142.3	600.4	77.7
623.4	112.6	141.9	600.7	77.6
628.4	112.3	141.6	601.0	77.5
633.4	112.8	142.2	600.5	77.2
638.4	112.5	141.8	600.8	77.1
643.4	112.2	141.4	601.1	76.9
648.4	112.7	142.0	600.6	79.1
653.4	112.4	141.6	600.9	79.0
658.4	112.1	141.2	601.3	78.9
663.4	112.5	141.8	600.8	78.5

Table 3.1-47 (Continued)
Double-Ended Pump Suction (SW Failure) Maximum Safeguards
Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
668.4	112.2	141.4	601.1	78.4
673.4	111.9	141.0	601.5	78.3
678.4	112.3	141.5	601.0	78.0
683.4	112.0	141.1	601.4	77.9
688.4	112.4	141.6	600.9	77.5
693.4	112.0	141.2	601.3	77.4
698.4	111.7	140.8	601.7	77.3
703.4	112.1	141.2	601.3	77.0
708.4	111.7	140.8	601.6	76.9
713.4	112.1	141.3	601.2	76.6
718.4	111.7	140.8	601.6	76.5
723.4	112.1	141.2	601.3	76.2
728.4	111.7	140.7	601.7	76.1
733.4	112.0	141.1	601.3	75.8
738.4	111.6	140.6	601.7	78.0
743.4	111.9	141.0	601.4	77.7
748.4	111.5	140.5	601.9	77.6
753.4	111.8	140.8	601.6	77.3
758.4	111.3	140.3	602.0	77.2
763.4	111.6	140.6	601.8	76.9
768.4	111.1	140.1	602.2	76.8
773.4	111.4	140.3	602.0	76.5
778.4	111.6	140.6	601.8	76.2

Table 3.1-47 (Continued)
Double-Ended Pump Suction (SW Failure) Maximum Safeguards
Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
783.4	111.1	140.0	602.3	76.1
788.4	111.3	140.2	602.1	75.8
793.4	111.4	140.4	601.9	75.5
798.4	110.9	139.8	602.4	75.4
803.4	111.0	139.9	602.3	75.1
808.4	111.2	140.1	602.2	74.9
813.4	111.3	140.2	602.1	76.9
818.4	111.3	140.3	602.0	76.6
823.4	110.8	139.6	602.6	76.5
828.4	110.8	139.7	602.5	76.2
833.4	110.8	139.7	602.5	76.0
838.4	110.8	139.7	602.5	75.7
843.4	110.8	139.6	602.6	75.5
848.4	110.7	139.5	602.6	75.2
853.4	110.6	139.4	602.7	75.0
858.4	111.0	139.9	602.3	74.6
863.4	110.8	139.7	602.5	74.4
868.4	110.6	139.4	602.7	76.4
873.4	110.9	139.7	602.5	76.1
878.4	65.8	82.9	647.5	87.6
1166.3	65.8	82.9	647.5	87.6
1166.4	63.7	80.1	649.6	84.4
1168.4	63.7	80.0	649.6	84.8

Table 3.1-47 (Continued)
Double-Ended Pump Suction (SW Failure) Maximum Safeguards
Post-Reflood Mass and Energy Releases (BVPS-2) (Case 2L1)

	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
1531.7	63.7	80.0	649.6	84.8
1531.8	53.2	61.2	660.2	21.8
2128.0	48.9	56.3	664.4	21.9
2128.1	52.9	60.9	781.1	92.2
3600.0	46.0	52.9	788.0	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)	1166.36(4)	1531.69(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	810.47	1071.08	2724.09
	Total Added	.00	.00	.00	130.69	810.47	1071.08	2724.09
Total Available		626.47	626.47	626.47	757.16	1436.95	1697.55	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	1310.57	1571.17	3224.18
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	417.78	417.78	630.78	1310.57	1571.17	3224.18
Total Accountable		626.47	626.46	626.46	748.31	1428.09	1688.70	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)	1166.36(4)	1531.69(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	26.75	35.35	194.25
	Decay Heat	.00	5.79	5.79	24.75	90.15	110.94	209.64
	Heat From Secondary	.00	.43	.43	.43	10.80	10.86	10.86
	Total Added	.00	6.22	6.22	29.49	127.70	157.14	414.74
Total Available		640.99	647.21	647.21	670.47	768.69	798.13	1055.73
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	2.94	2.71
	Primary Metal	115.85	109.61	109.61	88.61	49.41	43.26	40.35
	Secondary Metal	34.40	34.82	34.82	31.84	18.95	15.73	14.76
	Steam Generator	207.00	210.21	210.21	188.76	116.52	98.04	92.59
	Total Contents	640.99	388.24	388.24	342.31	217.04	189.15	179.58
Effluent	Break Flow	.00	258.48	258.48	320.12	543.60	577.74	845.81
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	258.48	258.48	320.12	543.60	577.74	845.81
Total Accountable		640.99	646.72	646.72	662.42	760.64	766.88	1025.40

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

Table 3.1-49 (Continued)
Double-Ended Pump Suction (SW Failure) Energy Balance
Maximum Safeguards (BVPS-2) (Case 2L1)

(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-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. 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 Section 4.0 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 BVPS-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 uprating 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 (ℓ / 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 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.

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 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 BVPS-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 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 power uprate 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 power uprate program. 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 power uprate 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 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 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 uprate 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 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 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 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 power uprate. 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 power uprate project. Table 3.3-1 presents the releases for the NSSS power uprate project.

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

power uprate. 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 power uprate 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. "BVPS-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

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)
1.0000	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
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 power uprate project.

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 5.0).

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 BVPS-2 at power uprate conditions with original steam generators. Since the CREA steam releases conservatively bound operation of BVPS-2 at 2910 MWT, they are expected to also bound operation of BVPS-1 at 2910 MWT with replacement steam generators.

4.0 CONTAINMENT ANALYSIS

4.1 Introduction

The MAAP5 code generalized containment model was used to assess the containment response to support the containment atmospheric conversion for both Beaver Valley units. This is a first time application of this model. The enhanced generalized containment model (GCM) includes treatment of momentum driven flows within the containment atmosphere, improved representation of the condensation heat transfer process, and a mechanistic representation for the entrainment of water films on walls and floors. The temperatures and pressures acting on the containment building and its supporting mechanical and electrical systems were analyzed to ensure that the containment parameters do not exceed the plant's design bases events acceptance criteria.

Those design bases events include 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. All these design bases events were addressed for the atmospheric containment configuration 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.

To ensure that the most conservative value of each of the above attributes was identified and evaluated the following considerations were addressed:

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 could be the maximum value or the minimum value. Thus it was necessary to select the most conservative value (max or min) of each input parameter for each attribute. 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 4.1.4.

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) (Unit 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 a main steam isolation valve (MSLB only),
- the failure of a main feedwater isolation valve (MSLB only), and
- 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.

A matrix was developed for each Beaver Valley unit to ensure the most conservative cases were identified and evaluated. These matrices are provided in Tables 4.1-1 and 4.1-2 as discussed in Section 4.1.4.

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 MAAP5 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 Chapter 6.

4.1.1 Application of MAAP5 to Containment Assessments

This section describes how the MAAP5 GCM is used for DBA evaluations. While the MAAP5 containment model can be used for other types of analyses, the extension of MAAP to realistic assessments with uncertainties as a design basis containment analysis tool is a new application of the MAAP code.

4.1.1.1 Parameter File/Nodalization

4.1.1.1.1 Nodalization

The application of the MAAP5 containment model to a commercial nuclear power plant begins with the characterization of the containment building geometry, emergency safeguard systems, etc., in the plant specific parameter file. While this parameter file includes specifications for the entire plant, including the reactor core and the reactor coolant system, the application of the MAAP5 containment model to design basis evaluations, with the external specification of the mass and energy releases into their containment, only requires that the containment information be qualified for the specific plant being evaluated. The remaining information can be used from a generic parameter file typical of the core and RCS design.

When formulating a parameter file, the most important decision lies in the specification of the number of nodes used to represent the containment. Considering the spectrum of containment building designs, it is difficult to be completely prescriptive in defining the nodalization scheme. However, there are a few rules that must be followed for DBA evaluations which can be combined with engineering judgment for the remaining features of the building.

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. With the different possible break locations, the containment nodalization must include the region surrounding the reactor coolant system, the loop compartment(s), and that the region above the operating deck where a MSLB would be postulated, 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. Similarly, main steam line breaks could be postulated in any one of the main steam lines above the operating deck and may also be postulated in another location if these steam lines extend downward along the containment wall before passing through the penetrations. If this is the case, the region near the containment wall should also be considered as a separate node since this could be a different postulated break location for this accident sequence type.

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 needs to 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 and/or hydrogen could accumulate. This is the only region of the Beaver Valley Containment model that uses multiple nodes to represent the given region.

Comparisons with large scale experiments (some open and some are highly compartmentalized) have shown that the number of nodes required to represent a large dry containment building is between 10 and 30 nodes depending upon the extent of compartmentalization inside the containment. In general, with the relatively open configuration above the operating deck and in the annular region, the number of nodes to be used to represent a typical large dry containment is

less than 20. The nodalization used for Beaver Valley is consistent with the guidance on nodalization in the MAAP5 Topical Report.

4.1.1.1.2 Flow Paths/Junctions

Once the nodalization is complete following the general rules outlined above, the junctions (flow paths) connecting the various nodes are also defined in the TOPOLOGY section of the parameter file. These junctions include those paths which are doorways, hatchways, open areas, grating, etc., as well as those flow paths which could be identified for the ventilation systems such as the Containment Fan Coolers (CFCs). 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 entry in the parameter file is very important in providing a realistic characterization, including the potential for global and countercurrent natural circulation, of the containment response to DBA conditions.

4.1.1.1.3 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 so slow that there are few heat sinks where the thermal conduction is developed through the entire width of the heat sink. Nonetheless the MAAP5 parameter file has the capability for all of these heat sinks to be identified as two-sided structures. This enables the same parameter file that is used for DBA evaluations to be used 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.

The LUMPED HEAT SINK section of the parameter can be used to represent those structures with negligible thermal resistance. Candidate structures are thin metal such as grating or HVAC ducting. Other uses could be to represent the instrument response, i.e., thermal measurements.

4.1.1.1.4 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. Issues related to the single failure criterion will be 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.

4.1.1.2 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. These include 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,
- containment temperatures compared to the Equipment Qualification (EQ) profile for equipment response,
- 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 both the short term discharges for these specific accident conditions which tend to 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 already 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 upon the temperature of the containment sump, since this represents that water which is eventually injected to the RCS, even though it may pass through a heat exchanger prior to this injection. Since the sump temperature changes with time, long term evaluations require feedback from the containment evaluation. Specifically, the mass and energy releases need the sump water temperature history such that long term analyses properly incorporate the decreasing temperature of the containment sump.

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 MAAP5 code. These input functions are used to incorporate the sump water temperature history (see above). These calculations are consistent with the Westinghouse methodology previously approved by the NRC. These user defined functions are characterized for 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 that are 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 containment cools. The user defined functions for the outlet specific enthalpies are as follows:

Unmixed: $h_{out,1} = (Q_{decay} + Q_{sg,1}) / W_1 + h_{in}$

$h_{out,2} = Q_{sg,23} / W_2 + h_{in}$

Mixed: $h_{out} = (Q_{decay} + Q_{sg,1} + Q_{sg,23}) / (W_1 + W_2) + h_{in}$

In these expressions, Q_{decay} is the core decay heat (Table 3.1-12), $Q_{sg,1}$ is the release rate of stored energy for the steam generator in the affected loop (from the upper head internals and upper elliptical heads), $Q_{sg,23}$ 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 recirculation water specific enthalpy 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 ensures 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 coverage value. 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.

4.1.1.3 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 and 16 psia, i.e., above and below the nominal atmospheric condition. Furthermore, the containment atmosphere temperature could be at its maximum value or its minimum value. These types of operating parameters have an influence on the specific attribute being evaluated, and the different boundaries of these operating conditions needs to be investigated to determine the set of conditions which maximizes the challenge to the attributes being evaluated. One means of organizing this set of conditions is to develop a parameter matrix that defines 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 Section 4.1.2. This matrix also identifies those issues related to uncertainties in the paint thickness on steel and concrete heat sinks, the effect 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 in a logical manner.

It is important to note that some of these evaluations may be insensitive to some 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 for example. Therefore, the focus of such a parameter matrix should be 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.

4.1.1.4 Uncertainty on the MAAP5 Modeling Parameters

The above discussion focuses on those uncertainties related to the containment operating envelope and these must also be combined with those uncertainties associated with the physical model parameters identified in the containment evaluation. As an example, the condensation model discussed in Chapter 4 of Reference 1 has an uncertainty related to the data that were used to formulate the empirical representation and its comparison with the spectrum of condensation experiments reported in the literature. These uncertainty boundaries need to be investigated with respect to the specific containment analysis being performed. For example, the condensation rate should be minimized for those evaluations which are focused on the possible peak containment pressure but would be maximized for those which are focused on the maximum liner temperature under design basis accident conditions. To ensure that these uncertainties are considered in a logical manner, these model parameters should also be controlled through a parameter matrix. Table 4.1-3 shows the way in which these model parameters can be tracked and applied for each type of attribute evaluation. In this regard, the minimum and maximum values are those defined through the FAI 5SSTAR process (Reference 1), and for the specific evaluations for containment, the parameters are reduced to two that are part of the containment evaluation.

The first is that related to the effective frictional forces on the containment walls, structures, etc., that decrease the momentum driven velocity created by the high pressure blowdown and spray operation following the accident initiation. Since the code uses Reynolds analogy to equate frictional losses and heat transfer, this tends to desensitize the evaluation with respect to the particular evaluation selected. Specifically, an increase in the parameter to increase the frictional losses also increases the heat transfer rate and vice-a-versa. As a result the integral response associated with the parameter has the effect of an increased frictional loss, which decreases the momentum induced velocities more quickly, also has an increased heat transfer rate during this decreasing flow. Therefore, the overall influence of the model parameter is reduced. The values for this increased frictional pressure drop associated with structure, liquid films on the walls and floors, additional structural members in the containment gas space such as hand railings, cables, gantry crane, R beams, etc., are assessed through evaluations with large scale tests. The values deduced from these numerous responses for the parameter FFMULT are defined in Reference 1.

The second, and the most influential of the MAAP5 modeling parameter is that associated with the condensation rate (FCOND). The specific uncertainty boundaries for this evaluation comes from the (MAAP5) 5SSTAR process which compares the condensation rate with the various experiments reported in the open literature. As discussed in the MAAP5 Topical Report

(Reference 1), the basic uncertainty boundaries were derived from the experiments reported by Dehbi et al, (Reference 3 and Reference 4) and these uncertainty boundaries were then compared with the experiments reported by Uchida et al, (Reference 6), Murase et al, (Reference 5) and Anderson et al, (Reference 2). With these comparisons, the specific values to be investigated for the condensation parameter are determined (see Reference 1).

These values are then controlled through a parameter matrix to assure that the particular values have been properly identified with the appropriate uncertainty boundary when analyzing a specific containment attribute.

4.1.2 Input Parameters and Assumptions

The input parameters for analysis are controlled through the combination of the run matrix and the parameter matrix. The run matrix has been developed for each unit and is illustrated in Tables 4.1-1 and 4.1-2. The parameter matrix that is developed for each of the types of analyses performed is given in Table 4.1-3. The parameter matrix controls the variations of the plant parameters (including the containment heat removal systems actuation signals and startup delay time) and those that require the specification of an uncertainty boundary. 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. In particular, 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. However, for the maximum sump temperature, the maximum enthalpy to the sump is established when the discharge from the two different sides of the break are added together. These are denoted in the input files as having the characterization of _MIX. This enables the information associated with the mass and energy releases to be clearly defined, since it is a modification from the manner in which the mass and energy releases were initially configured.

Another assumption which is used in evaluating the MSLB events is that the performance of the recirculation sprays was not credited in the licensing basis analyses for BVPS-2 as a sub-atmospheric containment. In this evaluation for the BVPS-1 and BVPS-2 responses as atmospheric containments, these spray systems are not credited in either the BVPS-1 or BVPS-2 MSLB evaluations for both peak pressure and peak temperature in the containment. To ensure that this is continually recognized in the model, the run designations for all of these MSLB events carry the qualifier of _NRS to indicate that there are no recirculation sprays being started in the evaluations.

4.1.3 Parameter Files for the Beaver Valley Units

One principal input element for the MAAP5 code is the parameter file which defines the containment geometry, nominal operating conditions, pump curves, etc. The MAAP5 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 Beaver Valley-specific modeling, detailed drawing and equipment specification reviews and site visits were conducted. The specific configurations of the BVPS-1 and BVPS-2 have been documented in unit-specific parameter files even though the differences between the two units are relatively small. This parameter file formulation has been assembled, reviewed, and documented in a report prepared in compliance with 10CFR50, Appendix B requirements and is the fundamental basis for the containment analysis.

Based on detailed drawing reviews and site visits and considering the plant-specific features of both Beaver Valley units, it was determined that each of the containments would most appropriately be represented with a 17 node model of the containment (see Figures 4.1-1, 4.1-2 and 4.1-3). This scheme enables the model to represent 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. Using multiple nodes above the operating deck enables the stratification of light gases to be calculated when this is part of the evaluation. The physical flow paths between the containment compartments are also included, as junctions, in the containment model. The junction areas and loss coefficients are based on the plant dimensions and are summarized in Table 4.1-4. MAAP5 calculates the quasi steady state nodal pressure distribution at each time step such that inertial effects due to flow acceleration are not required nor calculated due to the relatively slow containment pressurization (containment pressure benchmarks have demonstrated this behavior). Thus, inertial coefficients for each junction are not included in the parameter file.

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, their thicknesses 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, heat exchanger capacities, etc., are also contained in the unit-specific parameter file. The containment node volumes, metal heat sink areas and masses, and concrete heat sink areas and thicknesses included in the BVPS-1 and BVPS-2 containment model, are tabulated in Tables 4.1-5 and 4.1-6. The heat sinks include structural steel, concrete liners, ventilation ducts and supports, pipes, pipe supports and restraints, and heavy equipment.

The refueling cavity (node 14) in each Beaver Valley containment can hold up spray water during a LOCA. This 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 that interface with the reactor pressure vessel are removed. Thus, spray water must accumulate in each refueling cavity until it can overflow through the open hatches in the refueling seal ring. The water flows through the open hatches where it then accumulates in the reactor cavity (node 1) until it overflows the reactor cavity into 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. The holdup volume for BVPS-1 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 is actuated on a containment high-high pressure signal and after a start delay, directs cold water from the RWST to the quench spray ring header in containment. The nominal quench spray flow rate is 2916 gpm (BVPS-1) and 3000 gpm (BVPS-2) per train. For BVPS-1 only, 450 gpm of the quench spray pump discharge flow is diverted to the recirculation spray pumps suctions and the balance (2466 gpm) is directed to the spray header.

After a start delay, a recirculation spray (RS) system 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. The recirculation spray flow rate is 3500 gpm (BVPS-1) and 3450 gpm (BVPS-2) per train.

4.1.4 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 MAAP5 containment module with Beaver Valley-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 Beaver Valley-specific MAAP5 containment model. The MAAP5 containment model treated those Westinghouse generated mass and energy releases in a manner that resulted in conservative predictions of each of the containment attributes. For example, the mass and energy releases for large break LOCA events consider that the discharge from the postulated double-ended guillotine break are either separated steam-water streams into the containment, or a mixed stream depending upon which of these results is the maximum challenge to the containment. When evaluating containment pressure, the streams from the double-ended guillotine rupture large break are considered to be separate such that the maximum steam flow rate with the maximum enthalpy is discharged into the containment atmosphere. Conversely, when evaluating the behavior for the maximum sump temperature, these streams are mixed before being discharged to the containment to maximize the water temperature (enthalpy) being discharged to the containment sump.

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 4.1-1 and 4.1-2 for Units 1 and 2, respectively. These matrices identify the break size, the reactor power level, and the single active failure included in each sequence definition. LOCA breaks into each of the three loop compartments (nodes 5, 7, and 9) are included to evaluate possible asymmetries in the multiple node containment model. All of the MSLB breaks are considered in the upper compartment and those with the highest peak containment temperature are also considered for the annular containment compartments. This is to ensure that the bounding containment liner response is obtained. In these tables some of the MSLB cases indicate two active failures, i.e., one that 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 uncertainties related to features such as pump curves, heat exchanger capacities, model parameters, 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 and uncertainties 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 4.1-3.

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 (16.0 psia); conversely when evaluating the minimum NPSH available, this pressure should be taken as the minimum value (12 psia). The various limits listed in the table are used for the individual analyses that are being performed for the specific accident initiator. This parameter matrix also lists the boundaries of the specific model parameters for issues related to condensation, frictional pressure drop, etc., that control the important models in the MAAP5 code for energy transfer to the containment heat sinks. As indicated, these are generally set to the minimum values, since energy transferred to the structural and equipment heat sinks typically reduces the peak pressure, the peak temperature and increases the available NPSH. However, these minimum values are those developed through comparisons with experiments to ensure that they are consistent with the available experimental basis.

4.1.5 Uncertainty Discussion

Uncertainty evaluations include those aspects of the plant system which describe variations in the operating conditions, such as containment pressure, temperature, etc., prior to the onset of the accident condition. Moreover, these evaluations also include the uncertainties related to variabilities in the system response such as changes in the pump head curves, boundaries of the minimum and maximum heat removal capabilities in the heat exchangers, etc. Finally, the uncertainty boundaries related to the description of physical phenomena, such as condensation in the presence of noncondensible gases, momentum transfer from the break node to other nodes, etc., are as defined by the separate effects benchmarks. The separate effects benchmarks use tests which focus on a single physical phenomena rather than a large integral test which may involve multiple physical phenomena. The uncertainty analyses performed for these design basis events set the specific parameters of interest for a given analysis to the bounding values as described in the parameter matrix. Therefore, the approach is to always use the uncertainty boundary which results in the greatest challenge to the containment condition that were being evaluated.

4.1.6 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 temperature less than the equipment qualification curve,
- 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) used in the small break LOCA peak clad temperature (PCT) analysis (BVPS-1 only).

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 7.1.5 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 Units 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. Consequently, this pump 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. Hence, this BVPS-2 configuration has no flow path from the sump to the LHSI pump. 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 ft.

4.1.7 Results and Acceptability

4.1.7.1 Pressure and Temperature Response for LOCA

All of the Beaver Valley Design Bases LOCA events were investigated:

- 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),
- Small LOCA in reactor coolant system piping (SBA),
- Double ended rupture of main steam line inside containment (MSLB),

The large LOCA mass and energy release evaluations (see Chapter 3) determined that the DEHL and DEPS breaks bounded the DECL. A DEHL, DEPS or DECL break in a given RCS piping

loop would result in the mass and energy releases being directed to the same containment node, i.e., the loop compartment with the ruptured pipe. Thus, the containment nodalization would not discriminate the type of break. 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.

The sequence of events for the LOCA sequences for BVPS-1 are summarized in Tables 4.1-7 through 4.1-11 and for BVPS-2 in Tables 4.1-12 through 4.1-15.

Large break loss-of-coolant accidents have been analyzed for different break locations as well as different characteristics of the break. Specifically, the dilation 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 two independent blowdown streams from each side of the double-ended rupture as well as assuming that the two streams are thoroughly mixed before being discharged to the containment. Typically, the former is the mass and energy release which results in the maximum pressurization of the containment, whereas mixing of the two streams results in the maximum sump temperature history. Also, the DEHL break location results in the maximum containment pressure increase, whereas the DEPS break results in the highest containment sump temperature. Furthermore, since the DEHL produces the greatest containment pressurization, this break was considered to occur in each of the three different loop compartments (node numbers 5, 7, and 9) to ensure that the analysis included any differences with respect to which of the coolant loops experienced the design basis break condition.

The quench and recirculation spray behavior was included in these LOCA assessments. Only the quench sprays were considered in the MSLB results. The plant's spray header elevations, spray pump curves, and spray coverage for the multiple nodes were all modeled. The spray drop size range investigated in these containment assessments was 100 to 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 4.1-16 and 4.1-17 summarize the peak containment pressures for the large break LOCA cases identified in the run matrices for the two units. The reported pressures in these tables are in the break node. The pressures are reported as psig and referenced to an atmospheric pressure of 14.3 psi. As illustrated by these tables, the DEHL condition results in the maximum pressure increase. All of these sequences result in a pressurization which is less than the design basis value of 45 psig. This conclusion is valid for both the BVPS-1 and BVPS-2 evaluations. Furthermore, the evaluations demonstrate a small difference in the pressure increase depending upon which loop experiences the design basis DEHL, but the pressurization behavior is similar for all of the loop compartment evaluations. Figures 4.1-4 and 4.1-5 show the containment pressure history in the dome region, the gaseous atmosphere temperature in this locale, and the atmospheric temperature in the break room for the three different cases for both units. As

illustrated, there are only minor differences between the evaluations for the different break room locations.

As mentioned above, the acceptability of the results for the containment pressure increase is that the design basis break conditions analyzed using design basis methodology for the mass and energy releases to the containment must be less than the design basis structural capability of 45 psig. As demonstrated by the above analyses for various types of break conditions and mass and energy releases, these results meet the acceptance criteria of less then 45 psig for both Beaver Valley units.

4.1.7.2 SBLOCA Considerations

A spectrum of smaller breaks was also evaluated to determine the response of the containment depressurization system and impact to long term core cooling. The analysis conducted included breaks in both the cold leg and hot leg in sizes from 1 to 4 inches. The mass and energy input was derived utilizing the MAAP5 computer code. The derivation of the mass and energy input was consistent with that recently conducted for another plant.

The evaluation showed that for small break sizes, an increase in the containment high-high setpoint is necessary. This change delays initiation of the containment spray system to provide time for adequate sump inventory to accumulate to meet NPSH requirements for the recirculation spray pumps. This necessitated that a range be established for this setpoint in order to ensure that both adequate inventory was available in the sump for a SBLOCA and that peak containment pressure for a MSLB remained within the design limit.

All breaks in both the cold leg and hot leg will reach the CIB setpoint. However, a 2 inch break in the cold leg will not reach the CIB setpoint soon enough to prevent sump temperatures from exceeding safety injection limits. Cold leg breaks smaller than 2 inches will reach the CIB setpoint before sump temperatures exceed the safety injection limits. Cold leg breaks 3 inches and larger will also reach the CIB setpoint before the sump temperature exceeds the safety injection limits. The delay in the 2 inch cold leg break reaching the CIB setpoint results from phase separation which slows the containment pressurization for a period of time. All hot leg breaks will reach the CIB setpoint before sump temperatures exceed the safety injection limits.

Therefore, in order to support long-term cooling on ECCS recirculation following a SBLOCA, it will be necessary to manually initiate CIB prior to the RWST automatic transfer to the recirculation setpoint being reached. Guidance will be added to the Emergency Operating Procedures (EOPs).

4.1.7.3 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), the location of the steam line break, and the potential for the reactor to return to power as a result of the rapid cooldown caused by the single steam generator rapid depressurization. These are captured in the run matrix by considering different single failures, break locations, 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.

Using the design basis mass and energy releases discussed previously in this report, the containment pressure history was analyzed for each sequence assuming that the steam line break occurred in the region immediately above the operating deck where the steam lines exit the steam generator and then run horizontally toward the containment wall. For those accident sequences with the greatest pressurization, the break location is also analyzed for the annular region where the steam lines descend to the penetration through the containment wall. These particular runs are noted by an _N13, since this annular region is node 13 in the 17 node containment model.

Tables 4.1-18 and 4.1-19 are the peak pressure results for Beaver Valley Units 1 and 2, respectively. Included in these tables are those evaluations which also assume that the break location is in the annular region of node 13. As illustrated by these tables, 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 Beaver Valley 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 is done to be consistent with the previous licensing basis for BVPS-2 which did not credit recirculation sprays for any main steam line break condition.

4.1.7.4 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 temperature atmosphere for all 17 nodes in each sequence with the equipment qualification curve for each unit. These results are illustrated for those runs that resulted in the maximum temperature history in each unit in Figures 4.1-6 through 4.1-13. As illustrated in these figures, for BVPS-1 and BVPS-2, the containment temperature history is not always below the equipment qualification profile for all of the 17 nodes during the entire pressurization and depressurization condition. Refer to Section 7.10 for further discussion.

4.1.7.5 Peak Liner Temperature Response

The assessment of the peak liner temperature for the main steam line breaks (MSLB) was performed by examining the peak containment gas temperature for each of the seventeen nodes (17) for each of the MSLB sequences for both units. MSLB sequences were selected since they produced higher peak containment gas temperatures than the LOCA sequences. For the MSLB sequences with the highest gas temperature a family of liner temperatures for the upper and lower annular compartment plus the upper and lower dome regions are plotted against the liner and attachments temperature limit of 280°F. These results are illustrated for the sequences which resulted in the maximum temperature history in each unit in Figures 4.1-14 and 4.1-15. As illustrated in these figures, the liner temperature is below the upper bound limit of 280°F for the entire time. It should be noted that the MAAP5 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.

4.1.7.6 Subcompartment Pressurization Response

The subcompartment analyses addresses 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 uprate 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 power uprate. A similar evaluation was performed and determined that the current BVPS-2 UFSAR mass and energy releases are bounding for the power uprate 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 1) 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 2) have shown that peak ΔP is usually higher for a lower initial pressure of the subcompartment. The studies have also shown that a larger initial air mass in the subcompartment causes an increase in the predicted maximum ΔP. Increasing the compartment initial air pressure has the effect of increasing the air mass. However, the influence

of the initial pressure is greater than that of the initial air content. Therefore, the net effect of increasing the initial pressure will be a reduction of the predicted maximum ΔP.

References

1. Boyle, J. C. 1975: Meyer, S. P. 1981. Subcompartment Transient Response Code THREED, NU-092 (Proprietary).

2. NUREG/CR-1199 LA-8169-MS, Subcompartment Analysis Procedures, December 1979.

4.1.7.7 Combustible Gas Concentration Control and Monitoring

The post DBA hydrogen control systems recirculate the containment atmosphere through an electric recombiner following a LOCA in BVPS-1 or BVPS-2. Recombination ensures that there is no explosion or fire hazard due to hydrogen in the containment atmosphere following a DBA in accordance with 10CFR50.44. The combustible gas control system maintains the hydrogen concentration within the containment below 4-volume percent following a Design Basis Accident (DBA), which is the explosive mixture limit per Regulatory Guide 1.7, "Combustible Gas Concentration and Contamination Following Loss-of-Coolant Accident." The five major sources of hydrogen generation accounted for after a DBA are as follows:

1. Zirconium metal - water reaction involving 5% of the active core
2. Pressurizer gas space and RCS water
3. Radiolytic generation in the containment sump
4. Radiolytic generation in the reactor core
5. Corrosion of Aluminum and Zinc inside containment.

Conversion to an atmospheric containment does not result in an increase in any of the five major sources of hydrogen following a DBA. As described in Section 6.1.1, the impact of an atmospheric containment on the LOCA analysis is to increase the containment back pressure. This effect results in increased fuel cooling during reflooding and a lower calculated peak clad temperature (PCT) and Zr-H_2O reaction rates.

The post-DBA hydrogen control systems are Class 1E, and consist of two 100 percent design capacity thermal recombiners and associated motor-driven blowers, electric heaters, reaction chambers, and controls. Separate systems are provided for each unit. The blowers are used for drawing the containment atmosphere and passing it through the recombiners, in which the hydrogen and oxygen are combined to form water.

The two hydrogen recombiner units, located outside containment, in the safeguards area, are manually placed in operation following the DBA. Once in operation, the recombiners are designed to continue operating unattended.

Additional shielding is provided for the hydrogen recombiner control panels. This shielding together with the shielding provided by the containment structure ensures that an operator would be able to access the hydrogen recombiner panels during a post accident period of operation while limiting the radiation dose to an acceptable value, in accordance with NUREG-0737. Vital area access is discussed in detail in Chapter 5, Radiological Assessments.

With the change to an atmospheric containment, the hydrogen recombiner system will operate the same except that the mass flow rate through the recombiner units will increase. The increase causes an increase in recombiner blower load, but remains well within the design capacity of the blower. The increased air mass and available oxygen that results from the containment conversion does not change the stoichiometric basis for the existing limits, or the design capability of the recombiners. The increased mass flow rate results in improved recombiner performance over time, and lower long-term hydrogen concentrations inside containment.

The containment post-DBA hydrogen concentration was evaluated, using the currently licensed methodology, for operation at atmospheric conditions with an uprated power level. This new evaluation included the increased air mass in the atmospheric containment, and an increased margin on the actual quantities of corrodible materials. The evaluation is based on; the 5% maximum value of Zirconium fuel cladding reacting with the reactor coolant consistent with Regulatory Guide (R.G.) 1.7; and a 50 scfm (based on 68°F) recombiner blower flow rate. The results of this evaluation demonstrated that one recombiner unit in operation 13 hours following the DBA is sufficient to maintain the hydrogen concentration in the containment atmosphere below 4-volume percent.

4.1.7.8 Maximum Sump Temperature/Available NPSH

The results from the assessment of the peak sump temperature for the limiting LOCA sequence for 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 4.1-20.

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 combination of the decreasing sump temperature and the transient time of hot water flowing through the relatively long piping configuration, combined with the pipe wall thermal capacity results in a temperature transient which occurs over approximately one minute. To assess the average piping

temperature, which is important for assessing the thermal expansion, these DBA evaluations nodalized the piping arrangement for both the low head and high head injection systems axially along the coolant flow, as well as three nodes through the piping wall. The result of evaluating this transient and examining the average temperature of the piping system shows that the maximum average temperature is approximately 2°F less than the sump water temperature at the time that recirculation is started. As a result, the piping equilibrates to the sump temperature in approximately 1 minute after the onset of containment recirculation. The sump temperature approximately 1 minute after the onset of containment recirculation was used to specify the ECCS piping temperature for cold leg recirculation for both BVPS-1 and BVPS-2.

The maximum recirculation spray heat exchanger cooling water outlet temperatures for BVPS-1 were found to be 168.5/169°F (IRS/ORS) for Train A and 174.7/175.3 (IRS/ORS) for Train B and for BVPS-2 it was found to be 169°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 4.1-3.

The assessment of the available NPSH for the LOCA sequences was performed by comparing the calculated NPSH at the time of recirculation spray pump start and the switch to recirculation for the BVPS-1 LHSI pump 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 4.1-16 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 ft. Figure 4.1-17 illustrates that this NPSH requirement for the BVPS-2 RS system is satisfied. Consistent with the BVPS-2 licensing basis, only the sump saturated water temperature and no containment over pressure is used to quantify the RS pump NPSH. The NPSH results are summarized in Table 4.1-21 and Table 4.1-22.

4.1.7.9 Sensitivity Analyses

Each design basis evaluation necessitates that the uncertainties associated with physical processes and the spectrum of potential plant conditions to be evaluated and to determine the most challenging set of conditions related to the specific design basis attributes being evaluated. Many of these are straightforward. For example, evaluation of the peak containment pressure necessitates that the uncertainty boundary representing the minimum steam condensation rate is used, along with the maximum initial containment pressure for the spectrum of possible plant conditions. However, there are also numerous other parameters which have a very small influence on the particular attribute being investigated and in some cases, the specific variation providing the most challenging condition changes (but very little) with respect to the particular accident sequence being evaluated within a similar set of accident sequences. To address these very small influences, sensitivity analyses were performed to determine which of these

parameters would cause a significant change in the peak containment condition. Some of these parameters were found to be easily dealt with as bounding values such as the gap conductance between the containment carbon steel liner and the concrete wall. Both the minimum and maximum values were evaluated and it was found that the maximum value always provided the most limiting case and therefore was always used. Those variables which were found to have a very small influence on the containment response are documented in the sensitivity analysis and are indicated in the parameter matrix by (Table 4.1-3) footnote (1) to show that they have been evaluated but have no significant importance in the analyses.

4.1.7.10 Comparison of MAAP5 and LOCTIC Results

This section provides a comparison of the MAAP5 and LOCTIC containment analysis results. The containment loss of coolant accident (LOCA) and main steam line break (MSLB) evaluated all breaks listed in the run matrix provided in Chapter 4.1.2 for each Unit. Analysis of both Units is required because there are major differences in the design of the Emergency Core Cooling System and containment heat removal systems between the two Units. The LOCTIC code modeled the containment as one volume whereas MAAP5 modeled the containment with 17 volumes. The input data to both codes accurately describes the containment design including internal structures and engineered safety features.

The methodology used by LOCTIC for the results given in this section differs from the usual methodology for safety analysis in two ways. The first concerns the assumed flash model for the break effluent. The Safety Analysis Report methodology assumes that both the steam and the liquid phases of the break effluent come to temperature equilibrium with the containment atmosphere. This is the temperature flash assumption. Relaxation of the temperature flash assumption was assessed by selecting LOCTIC's pressure flash (PF) option. This option assumes that vaporization of the break effluent liquid ceases when the liquid temperature is cooled to the saturation temperature corresponding to the containment total pressure by flashing. The liquid flows to the containment floor at this temperature and the vapor effluent is added to the containment atmosphere. The pressure flash model is more characteristic of the flashing phenomenon than the temperature flash model. The containment peak pressure after a reactor coolant pipe double-ended rupture (DER) occurs quickly (less than 20 seconds) after the break. The temperature flash assumption requires time for the liquid phase to cool by an evaporative (slow) process before reaching the floor. Since the time available for evaporative cooling of the liquid phase that remains after the flashing is short, the liquid phase is expected to drop to the floor at an elevated temperature compared to the containment atmosphere temperature. LOCTIC shows that use of the pressure flash reduces the peak containment pressure by more than 2 psi.

The second concerns the condensation heat transfer coefficient for the passive heat sinks. For the loss of coolant accident, the Safety Analysis Report methodology assumes one times the Tagami correlation for the peak condensation coefficient. The impact of increasing the condensation heat transfer coefficient on the containment heat sinks in addition to assuming a pressure flash of the break effluent was also investigated. Since the NRC Staff also requires a multiplier of 4 on the

Tagami correlation for the calculation of the minimum containment pressure in accordance with 10 CFR 50, Appendix K, the uncertainty imposed is rather large +/- 60% of the best-estimate (i.e., average) value of 2.5 times Tagami. The typical uncertainty for heat transfer coefficients is +/-20%. Using a multiplier of 2 on the correlation essentially lowers the uncertainty to -20% on the low end. The LOCTIC results show that the peak containment pressure is below the containment design pressure by almost 2 psi when the pressure flash is employed and a multiplier of 2 is applied to the Tagami peak condensing correlation.

The containment pressure, temperature, and ESF pump available NPSH results for MAAP5 and LOCTIC are compared below.

Loss of Coolant Accident

Peak Containment Pressure Results

The peak containment pressure results from a hot leg double-ended rupture (HLDER). Both MAAP5 and LOCTIC show this to be the limiting break. The containment peak pressures calculated by the two codes are listed in the following table.

Peak Containment LOCA Pressure Comparison		
Code/Option	**Break**	**Peak Pressure, psig**
BVPS-1		
LOCTIC/PF+Tagami x 2	8L	43.3
MAAP5	8L*	42.5
BVPS-2		
LOCTIC/PF+Tagami x 2	3L	44.7
MAAP5	3L*	43.4

* Break into node #7.

Peak Containment Temperature Results

The peak post-LOCA containment atmosphere temperature also occurs after an HLDER. A comparison of the MAAP5 and LOCTIC results is presented in the following table.

Peak Containment LOCA Temperature Comparison		
Code/Option	**Break**	**Peak Temperature, °F**
BVPS-1		
LOCTIC/PF+Tagami*2	8L	265.9
MAAP5	8L	292.7
BVPS-2		
LOCTIC/PF+Tagami*2	3L	267.9
MAAP5	3L	295.9

Low Head Safety Injection Pump Available NPSH

The available net positive suction head (NPSH) following a pump suction DER with minimum engineered safety features is compared for the 6L break in BVPS-1. The results calculated by the MAAP5 and LOCTIC codes are compared in the following table.

Minimum Available NPSH for LHSI Comparison		
Code/Option	**Break**	**Minimum NPSHA, ft**
LOCTIC/PF+Tagami x 2	6L	19.4
MAAP5	6L	23.7

Recirculation Spray Pump Available NPSH

The minimum available NPSH for the recirculation spray pumps occurs following a HLDER with no single active failure. The MAAP5 and LOCTIC predictions of the minimum available NPSH for the inside and outside recirculation spray pumps are provided in the following table.

Minimum Available NPSH for IRS/ORS Comparison		
Code/Option	**Break**	**Minimum NPSHA, ft**
BVPS-1		
LOCTIC/PF+Tagami x 2	2L	11.8/11.1
MAAP5	2L	16.1/16.8

Main Steam Line Break

Peak Containment Pressure Results

The peak containment pressure for the BVPS-1 occurs following a 1.4 ft^2 DER with an MSCV failure at 30 percent power. The peak containment pressure for BVPS-2 occurs following a 1.069 ft^2 DER with a MSIV failure at zero power. The MAAP5 and LOCTIC results for this break are compared in the following table.

Peak MSLB Containment Pressure Comparison		
Code/Option	**Break**	**Peak Pressure, psig**
BVPS-1		
LOCTIC/Uchida x 2	15M	44.5
MAAP5	15M*	44.9
BVPS-2		
LOCTIC/Uchida x 2	17M	44.6
MAAP5	16M*	44.9

* Break into node #13.

Peak Containment MSLB Temperature Results

For BVPS-1 the maximum MSLB temperature results from a 1.4 ft^2 DER with an MSCV failure at 100.6 percent power. For BVPS-2 the maximum temperature results from a 1.069 ft^2 DER with an CIB failure at 100.6 percent power. The MAAP5 and LOCTIC results for this break are compared in the following table.

Peak Containment MSLB Temperature Comparison		
Code/Option	**Break**	**Peak Temperature, °F**
BVPS-1		
LOCTIC/Uchida x 2	3M	358.2°F
MAAP5	3M	321.1°F
BVPS-2		
LOCTIC/Uchida x 2	2M	347.2°F
MAAP5	2M	307°F

MAAP5/LOCTIC Comparison Summary

As the above comparisons show, the one node LOCTIC results and the 17-node MAAP5 results are quite similar when the one node model uses improved break effluent flashing and condensation heat transfer coefficients assumptions.

4.1.7.11 Summary of Containment MAAP5 Results

The results for the containment analyses for the atmospheric configuration have been compared to the applicable acceptance criteria for both Units 1 and 2. The results for each attribute were calculated using the applicable set of linearly (not statistically) summed bounding parameters and uncertainties and 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 Beaver Valley 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. The impact of the power uprate on subcompartment pressurization was found to result in differential pressures on the compartment walls that provide sufficient margin for their structural design. The combustible gas control evaluation demonstrated that one recombiner unit in operation is sufficient to maintain the hydrogen concentration in the containment atmosphere below the acceptance criterion of 4-volume percent.

4.1.7.12 References

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2. Anderson, M. A. et al., 1998 “Experimental Analysis of Heat transfer Within the AEP 600 Containment Under Postulated Accident Conditions,” Nuclear Engineering and Design, 185, pp. 153-172.

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4. Dehbi, A. A. et al, 1991b, “Condensation Experiments in Steam-Water and Steam-Air-Helium Mixtures Under Turbulent Natural Convection,” AIChE Symposium Series, Heat Transfer, Volume 87, pp. 19-28.

5. Murase, M. et al., 1992, “Evaporation and Condensation Heat Transfer with a Noncondensible Gas Present,” Japan-US Seminar on Two-Phase Flow Dynamics, Berkeley, California, July 5-11, pp. 415-424.

6. Uchida, H. et al., 1965, “Evaluation of Post-Accident Cooling Systems of Light-Water Power Reactors,” Proceedings of the International Conference on Peaceful Uses of Atomic Energy, Geneva, Switzerland, Volume 13, pp. 93-102.

Table 4.1-1
MAAP5 Run Matrix for Containment Analysis Beaver Valley – BVPS-1

Break Description	Power Level, %	Single Failure	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	Max. Sump Temp.(1)/Available NPSH			Maximum Service Water Outlet Temp.
						IRS	ORS	LHSI	
Loss-Of-Coolant Accident									
1L-DEHL MIN SI (unmixed)	100.6	DG				X	X		
1L-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	X	
4L-DEHL MAX SI (mixed)	100.6	None				X	X	X	
5L-DEPS MIN SI (mixed)**	100.6	DG				X	X	X	
6L-DEPS MIN SI	100.6	DG	X	X	X	X	X	X	X
7L-DEPS MAX SI	100.6	CIB	X	X	X	X	X	X	
7L1-DEPS MAX SI	100.6	SW**						X	

Table 4.1-1 (Continued)
MAAP5 Run Matrix for Containment Analysis Beaver Valley – BVPS-1

Break Description	Power Level, %	Single Failure	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	Max. Sump Temp.(1)/Available NPSH			Maximum Service Water Outlet Temp.
						IRS	ORS	LHSI	
8L-DEHL (break into node #5)	100.6	None	X	X					
8L-DEHL (break into node #7)	100.6	None	X	X					
8L-DEHL (break into node #9)	100.6	None	X	X					
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					
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					

Table 4.1-1 (Continued)
MAAP5 Run Matrix for Containment Analysis Beaver Valley – BVPS-1

Break Description	Power Level, %	Single Failure	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	Max. Sump Temp.(1)/Available NPSH			Maximum Service Water Outlet Temp.
						IRS	ORS	LHSI	
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					
16M-1.4 ft^2 DER	30	CIB/MFIV	X	X					
17M-0.424 ft^2 Split	30	MSCV	X	X					
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				

Table 4.1-1 (Continued)
MAAP5 Run Matrix for Containment Analysis Beaver Valley – BVPS-1

Break Description	Power Level, %	Single Failure	Peak Pressure	Peak/EQ Temperature	Peak Liner Temperature	Max. Sump Temp.(1)/Available NPSH			Maximum Service Water Outlet Temp.
						IRS	ORS	LHSI	
23M-0.209 ft^2 Split	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)

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.

* All MSLBs will assume the break to be into node #15. Those with the highest temperatures will also be assumed to break into the annular region nodes, i.e., node 13.

** This case (SW failure) is enveloped by Case 7L (CIB failure). Therefore, this break's results were not reported.

Table 4.1-2
MAAP5 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 (break into node #5)	100.6	(2)	X	X		X	
3L-DEHL (break into node #7)	100.6	(2)	X	X		X	
3L-DEHL (break into node #9)	100.6	(2)	X	X		X	
Main Steam Line Break*							
1M-4.6 ft^2 DER	100.6	MSIV/CIB	X	X			
2M-4.6 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			

Table 4.1-2 (Continued)
MAAP5 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.
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		
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			
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			

Table 4.1-2 (Continued)
MAAP5 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.

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.

* All MSLBs will assume the break to be into node #15. Those with the highest temperatures will also be assumed to break into the annular region nodes, i.e., node 13.

Table 4.1-3
Containment Parameter Limits and Uncertainty Boundaries 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[1]	Max	Min[c]	Max	Max
Initial containment temperature	Max[e]	Max	Max	Max	Max	Max
Initial containment relative humidity	Min[f]	Max[g]	Max	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]	Max	Max	Max
Effective heat transfer coefficient for the paint on the carbon steel	Min[j]	Min	Max[d]	Min	Min	Min
Paint thickness on concrete heat sinks	Max[k]	Max	Max	Max	Max	Max

Table 4.1-3 (Continued)
Containment Parameter Limits and Uncertainty Boundaries 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
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
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[p]	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

Table 4.1-3 (Continued)
Containment Parameter Limits and Uncertainty Boundaries 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
Start delay for quench spray	Max[bb]	Max	Max	Min[cc] – IRS/ORS Max – LHSI	Max	Max
Quench spray flow rate	Min[jj]	Min	Min	Min[dd] –LHSI Max[kk] – RS	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]	N/A[3]	N/A[4]	Min[q] Max for IRS & ORS[ii]	Min	Max[ee]
Recirculating spray flow rate	N/A[2]	N/A[3]	N/A[4]	Min[r] – LHSI Max[ff] – RS	Min	N/A
Recirculation HX cooling water temperature	N/A[2]	N/A[3]	N/A[4]	Min[z]	Max	Max
Recirculation spray HX cooling water flow rate	N/A[2]	N/A[3]	N/A[4]	Min[t] – LHSI Max[gg] - RS	Min	Min
Range of usable RWST volume prior to switchover	N/A[2]	N/A[3]	N/A[4]	Min[u]	Min	Min

Table 4.1-3 (Continued)
Containment Parameter Limits and Uncertainty Boundaries 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
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)
MAAP5 Model Parameters						
Quench spray droplet diameter	Max[v]	Max	Max	Min[w]	Min	Min
Multiplier for the wall frictional forces which directly influences the heat transfer coefficient through Reynolds analogy[x]	Min	Min	Min	Min/Max[1]	Max	Max
Multiplier for the condensation heat transfer coefficient on vertical and downward facing surfaces with respect to the standard heat-mass transfer analogy[x]	Min	Min	Max	Min	Min	Min

Table 4.1-3 (Continued)
Containment Parameter Limits and Uncertainty Boundaries 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
Fraction of the water break flow remaining airborne as suspended water(x)	Min	Min	Min	Min	Min	Min

(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) 16 psia

(c) 12 psia

(d) 2667 BTU/hr/ft^2/ºF

(e) 105°F

(f) 50%

(g) 100%

(h) 100 BTU/hr/ft^2/°F

(i) 0.009 inches (BVPS-1) and 0.0105 inches (BVPS-2)

(j) 500 BTU/hr/ft^2/°F (BVPS-1) and 400 BTU/hr /ft^2/ºF (BVPS-2)

(k) 0.008 inches (BVPS-1) and 0.020 and 0.030 inches for the containment walls/ceilings and floors respectively. (BVPS-2)

(m) 0.026 in

(n) 65°F

Table 4.1-3 (Continued)
Containment Parameter Limits and Uncertainty Boundaries 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
(o) BVPS-1 used 40.0 psia (max) and 37.4 psia (min); BVPS-2 limiting pressure case used 36.7 psia (max) and 34.1 psia (min).						
(p) 22 psia (max); 18 psia (min)						
(q) 2.156E6 BTU/hr/°F (BVPS-1) and 3.3 x 10^6 BTU/hr/°F (BVPS-2)						
(r) 3500 gpm (BVPS-1) and 3450 (BVPS-2); runout flow per 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 403,000 gals (BVPS-2)						
(v) 1 mm						
(w) 0.1 mm						
(x) The Min and Max for this parameter are defined through the FAI 5SSTAR process in topical report, WCAP-15844.						
(y) 24 psia (max); 20 psia (min)						
(z) 32ºF						
(aa) 250 BTU/hr/ft^2/°F (BVPS-1) and 50/75 BTU/hr/ft^2/°F (Min/Max BVPS-2) floor/walls and ceilings.						
(bb) 85 seconds (BVPS-1) and 85.5 seconds (BVPS-2)						
(cc) 64 seconds (BVPS-1) and 58 seconds (BVPS-2)						
(dd) 300 seconds (BVPS-1) and 720 seconds (BVPS-2)						

Table 4.1-3 (Continued)
Containment Parameter Limits and Uncertainty Boundaries 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
(ee) 2.976E6 BTU/hr/ºF (BVPS-1) and 4.35E6 BTU/hr/ºF (BVPS-2)						
(ff) 3500 gpm (BVPS-1) and 3450 gpm (BVPS-2); runout flow per pump curve						
(gg) 4500 gpm (BVPS-1) and 5500 gpm (BVPS-2)						
(hh) Min accumulator tank volume: 1436 ft^3 (BVPS-1) and 1444 ft^3 (BVPS-2)						
(ii) BVPS-1 only						
(jj) 2835 gpm (BVPS-1) which includes 450 gpm of enhancement flow and 3000 gpm (BVPS-2)						
(kk) 2916 gpm (BVPS-1) which includes 450 gpm of enhancement flow and 3000 gpm (BVPS-2)						
(1) This is analyzed both ways since the difference is small and the limiting case in not clear a priori.						
(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 peak temperature since sprays start after the peak value.						
(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 MAAP5 when containment response is quantified given Westinghouse supplied mass and energy releases are used.						

Table 4.1-4
Beaver Valley MAAP5 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	2	10.31	1.83
2	1	5	6.0	1.94
3	1	7	6.0	1.94
4	1	9	6.0	1.94
5	2	4	417.0	2.85
6	2	5	160.0	2.49
7	2	6	33.5	2.61
8	2	7	155.0	2.49
9	2	8	976.0	0.01
10	2	9	140.6	2.49
11	2	10	1166.0	0.25
12	2	11	1166.0	0.25
13	3	5	28.0	2.35
14	13	12	903.2	0.01
15	4	5	28.0	2.35
16	4	10	517.8	2.83
17	5	6	56.0	2.35
18	5	15	384.8	1.87
19	6	7	56.0	2.35
20	6	11	56.0	2.35
21	6	15	20.0	1.63
22	7	15	389.4	1.93
23	8	11	800.0	2.76
24	8	15	673.4	0.25

Table 4.1-4 (Continued)
Beaver Valley MAAP5 Parameter File Summary of Junction Flow Areas

Junction Number	Upstream Node Number	Downstream Node Number	Junction Flow Area (ft^2)	Junction Loss Coefficient
25	9	10	56.0	2.35
26	9	15	389.4	1.93
27	11	10	493.3	0.01
28	10	12	2514.5	0.25
29	11	13	2514.5	0.25
30	12	15	792.3	2.72
31	13	15	130.0	2.68
32	17	16	7980.0	0.01
33	16	15	8120.0	0.01
34	16	13	1822.0	0.01
35	16	12	1822.0	0.01
36	15	14	1105.0	0.01
37	14	2	0.0 (Refueling canal drain path)	0.75
38	14	1	23.38 (BVPS-1); 20.04 (BVPS-2)	2.45

Table 4.1-5
Beaver Valley MAAP5 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	265,732	33,539
11	Lower annulus north (BVPS-1) or south (BVPS-2) half	85663	85663	249,867	31,305
12	Upper annulus south (BVPS-1) or north (BVPS-2) half	80082	80082	114,365	16,192
13	Upper annulus north (BVPS-1) or south (BVPS-2) half	80294	80294	193,250	17,152
14	Refueling cavity	33734	36620	131,960	5,522
15	Upper compartment cylindrical section	347071	347071	481,486	15,339
16	Upper compartment lower dome region	416,601	416,601	583,731	34,062
17	Upper compartment upper dome region	108,635	108,635	0	0
	TOTAL	1,756,598	1,761,508	4,003,198	254,203

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.

Table 4.1-6
Beaver Valley MAAP5 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(1)	4.00	958	2	1,915
3	Shield wall to loop C	4.50	195	2	389
4	Refuel cavity wall to PZR(1)	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(1)	4.00	826	2	1,653
10	Reactor cavity floor (2)	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(2)	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

Table 4.1-6 (Continued)
Beaver Valley MAAP5 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
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
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(2)	4.50	8,432	1	8,432
43	Lower annulus north half outer wall(2)	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(2)	4.50	7,569	1	7,569

Table 4.1-6 (Continued)
Beaver Valley MAAP5 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
46	Upper annulus north half crane wall	2.75	9,038	2	18,076
47	Upper annulus north half outer wall(2)	4.50	7,569	1	7,569
48	Fuel transfer canal floor(1)	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
59	Refuel cavity wall to upper annulus south half(1)	4.00	1,033	2	2,066
60	Refuel cavity wall to upper annulus north half(1)	4.00	1,181	2	2,361
61	Containment shell sections with embedment plates in lower compartment(2)	4.50	338	1	338
62	Containment shell sections with embedment plates in lower south annulus(2)	4.50	492	1	492
63	Containment shell sections with embedment plates in lower north annulus(2)	4.50	492	1	492

Table 4.1-6 (Continued)
Beaver Valley MAAP5 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
64	Containment shell sections with embedment plates in upper south annulus(2)	4.50	525	1	525
65	Containment shell sections with embedment plates in upper north annulus(2)	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	735	1	735
72	Unlined portion of lower annulus south half outer wall	4.50	960	1	960
73	Unlined portion of lower annulus north half outer wall	4.50	960	1	960
74	Unlined portion of upper annulus south half outer wall	4.50	983	1	983
75	Unlined portion of upper annulus north half outer wall	4.50	983	1	983
	TOTAL				217,752

(1) Includes unpainted stainless steel liner and gap resistance between liner and concrete.

(2) Includes painted carbon steel liner and gap resistance between liner and concrete.

Table 4.1-7 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)
4.8	Containment High-High Setpoint is reached
11.9	Broken Loop Accumulator Begins Injecting Water
12.1	Intact Loop Accumulator Begins Injecting Water
16.1	Peak Containment Pressure During Blowdown
22.2	End of Blowdown Phase

Table 4.1-8
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
1.0	Containment High-High Setpoint is reached
4.8	Low Pressurizer Pressure SI Setpoint is reached (1745 psia)
12.7	Broken Loop Accumulator Begins Injecting Water
12.9	Intact Loop Accumulator Begins Injecting Water
17.4	Peak Containment Pressure During Blowdown
27	End of Blowdown Phase
27	Safety Injection Begins
56.5	Accumulator Water Injection Ends
86.6	Quench Spray is initiated
222.2	End of Reflood Phase
301	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 temperature, and hydrogen recombiner assessments which may require 1 day or 30 day transients.

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 4.1-9
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)
4.8	Containment High-High Setpoint is reached
12.7	Broken Loop Accumulator Begins Injecting Water
12.9	Intact Loop Accumulator Begins Injecting Water
16.1	Peak Containment Pressure During Blowdown
21.0	End of Blowdown Phase
27	Safety Injection Begins
56.9	Accumulator Water Injection Ends
86.5	Quench Spray is initiated
215.3	End of Reflood Begins
301.2	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 temperature, and hydrogen recombine assessments which may require 1 day or 30 day transients.

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 4.1-10
BVPS-1 Sump Temperature 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 4.1-11
BVPS-1 Sump Temperature 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 4.1-12 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 (1745 psia)
4.1	Containment High-High Setpoint is reached
11.5	Broken Loop Accumulator Begins Injecting Water
11.6	Intact Loop Accumulator Begins Injecting Water
17.9	Peak Containment Pressure During Blowdown
19.2	End of Blowdown Phase

Table 4.1-13
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)
4.3	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
87.4	Quench Spray is initiated
214.2	End of Reflood Phase
721.3	Recirculation Spray is initiated
2948	ECCS Recirculation Begins
3600[(1)]	Transient Modeling Terminated

(1) Except for long term attributes such as EQ profiles, sump water temperature, and hydrogen recombiner assessments which may require 1 day or 30 day transients.

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 4.1-14
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)
4.3	Containment High-High Setpoint is reached
12.9	Broken Loop Accumulator Begins Injecting Water
13.0	Intact Loop Accumulator Begins Injecting 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 Phase
721.3	Recirculation Spray is initiated
2782	ECCS Recirculation Begins
3600[(1)]	Transient Modeling Terminated

(1) Except for long term attributes such as EQ profiles, sump water temperature, and hydrogen recombiner assessments which may require 1 day or 30 day transients.

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 4.1-15
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 Begins
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 water temperature, and hydrogen recombiner assessments which may require 1 day or 30 day transients.

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 4.1-16
MAAP5 Containment Peak Pressure Results for a Design Basis Large Break LOCA
Beaver Valley – BVPS-1

Description	Power Level, %	Single Failure	Peak Pressure (psig) (3)	Pressure at One Hour (psig)
1L-DEHL MIN SI(1)	100.6	DG	36.9	2.8
1L-DEHL MIN SI(1)	100.6	LHSI	36.9	-0.7
2L-DEHL MAX SI(1)	100.6	None	36.9	1.1
3L-DEHL MIN SI(1)	100.6	DG	36.9	1.9
4L-DEHL MAX SI(1)	100.6	None	36.9	0.7
6L-DEPS MIN SI	100.6	DG	41.8	9.0
6L-DEPS_MIX MIN SI	100.6	DG	39.5	7.0
7L-DEPS MAX SI	100.6	CIB	41.0	9.2
7L-DEPS_MIX MAX SI	100.6	CIB	41.0	8.3
8L-DEHL (break into node #5)	100.6	None	42.2	(2)
8L-DEHL (break into node #7)	100.6	None	42.5	(2)
8L-DEHL (break into node #9)	100.6	None	42.1	(2)

Single Failures – Failed Equipment

CIB One train each, QSS, RSS

DG One train each, SI, QSS, RSS

LHSI One LHSI train

(1) Parameter values set to maximize calculated sump temperature not peak pressure attribute per the run matrix in Table 4.1-1.

(2) Only blowdown (19.2 seconds) was quantified. Peak pressure occurs before any active failure could occur.

(3) Gauge pressure is referenced to 14.3 psi atmospheric pressure. Reported peak pressure is for break node.

Table 4.1-17
MAAP5 Containment Peak Pressure Results for a Design Basis Large Break LOCA
Beaver Valley – BVPS-2

Description	Power Level, %	Single Failure	Peak Pressure (psig)(1)	Pressure at One Hour (psig)(1)
1L-DEPS MIN SI	100.6	DG	41.1	6.3
2L-DEPS MAX SI	100.6	CIB	41.1	6.4
3L-DEHL (break into node #5)	100.6	None	43.3	(2)
3L-DEHL (break into node #7)	100.6	None	43.4	(2)
3L-DEHL (break into node #9)	100.6	None	43.2	(2)

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. Reported pressure is for break node.

(2) Only blowdown (22.2 seconds) was quantified. Peak pressure occurs before any active failure could occur.

Table 4.1-18
MAAP5 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	34.8
2M-4.6 ft^2 DER	100.6	CIB/MFIV	28.6
3M-1.4 ft^2 DER	100.6	MSCV	41.1
4M-1.4 ft^2 DER	100.6	CIB/MFIV	32.1
5M-0.725 ft^2 Split	100.6	MSCV	37.0
6M-0.725 ft^2 Split	100.6	CIB/MFIV	34.3
7M-4.6 ft^2 DER	70	MSCV	37.1
8M-4.6 ft^2 DER	70	CIB/MFIV	30.8
9M-1.4 ft^2 DER	70	MSCV	43.2
10M-1.4 ft^2 DER	70	CIB/MFIV	35.1
11M-0.710 ft^2 Split	70	MSCV	37.8
12M-0.710 ft^2 Split	70	CIB/MFIV	35.6
13M-4.6 ft^2 DER	30	MSCV	38.4
14M-4.6 ft^2 DER	30	CIB/MFIV	31.9
15M-1.4 ft^2 DER	30	MSCV	44.8
15M_N13-1.4 ft^2 DER	30	MSCV	44.9
16M-1.4 ft^2 DER	30	CIB/MFIV	36.6
17M-0.424 ft^2 Split	30	MSCV	37.5
18M-0.424 ft^2 Split	30	CIB/MFIV	35.6
19M-4.6 ft^2 DER	0	MSCV	35.1
20M-4.6 ft^2 DER	0	CIB/MFIV30	31.7
21M-1.4 ft^2 DER	0	MSCV	42.4
22M-1.4 ft^2 DER	0	MFIV30	37.3

Table 4.1-18 (Continued)
MAAP5 Peak Pressure Results for a Design Basis Main Steam Line Break*
Beaver Valley – BVPS-1

Description	Power Level, %	Single Failure	Peak Pressure (psig)
22M_N13-1.4 ft^2 DER	0	MFIV30	37.6
22M1-1.4 ft^2 DER	0	CIB	36.0
23M-0.209 ft^2 Split	0	MSCV/MFIV30	35.6
24M-0.209 ft^2 Split	0	CIB/MFIV30	36.0

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)

* All MSLBs specify the break to be into node #15. Those with the highest temperatures are also analyzed for the same break in the annular region nodes, i.e., node 13. These cases are designated by _N13 in the case description. Also, the mass and energy releases for all these cases have been analyzed with a water temperature of 105°F in the RWST. This results in a conservative (overestimate) of the maximum containment pressure.

Table 4.1-19
MAAP5 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	37.6
2M-1.069 ft^2 DER	100.6	MFIV/CIB	37.3
3M-0.753 ft^2 Split	100.6	CIB	33.7
4M-0.753 ft^2 Split	100.6	MSIV	34.4
5M-0.753 ft^2 Split	100.6	MFIV	33.6
6M-1.069 ft^2 DER	70	MSIV	38.7
7M-1.069 ft^2 DER	70	MFIV/CIB	38.0
8M-0.757 ft^2 Split	70	CIB	35.4
9M-0.757 ft^2 Split	70	MSIV	35.7
10M-0.757 ft^2 Split	70	MFIV	34.6
11M-1.069 ft^2 DER	30	MSIV	41.1
12M-1.069 ft^2 DER	30	MFIV/CIB	39.7
13M-0.756 ft^2 Split	30	CIB	38.1
14M-0.756 ft^2 Split	30	MSIV	38.3
15M-0.756 ft^2 Split	30	MFIV	36.7
16M-1.069 ft^2 DER	0	MSIV	**44.8
16M_N13-1.069 ft^2 DER	0	MSIV	**44.9
17M-1.069 ft^2 DER	0	MFIV/CIB	43.0
17M_N13-1.069 ft^2 DER	0	MFIV/CIB	43.0
18M-0.608 ft^2 Split	0	CIB	41.6

Table 4.1-19 (Continued)
MAAP5 Peak Pressure Results for a Design Basis Main Steam Line Break*
Beaver Valley – BVPS-2

Description	Power Level, %	Single Failure	Peak Pressure (psig)
19M-0.608 ft^2 Split	0	MSIV	40.4
20M-0.608 ft^2 Split	0	MFIV	39.1

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

* All MSLBs will assume the break to be into node #15. Those with the highest temperatures were also assumed to break into the annular region nodes, i.e., node 13. These are identified with (_N13) in the description. Also, the mass and energy releases for all these cases have been analyzed with a water temperature of 105°F in the RWST. This results in a conservative (over-estimate) of the maximum containment pressure.

** These two sequences were analyzed using a maximum spray setpoint of 36.7 psia.

Table 4.1-20
MAAP5 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
BVPS-1				
1L-DG	248°F	239°F	173.2° F	145°F
1L-LHSI	248°F	233.5°F	130.2°F	N/A
2L	248°F	240.5°F	162.7°F	N/A
3L(1)	248°F	251°F	177.7°F	145°F
4L	248°F	244.5°F	142.5°F	N/A
6L	248°F	247.4°F	184.2°F	145°F
7L	248°F	247.5°F	178.2°F	145°F
BVPS-2				
1L	248°F	247°F	109.2°F	99.3°F
2L	248°F	244°F	108.7°F	99.4°F
2L1	248°F	236°F	167.7°F	131°F
2L2	248°F	236.5°F	111°F	N/A
3L	248°F	211°F	117°F	99°F

(1) Per the run matrix, the results of this case are provided for completeness. However, they should not be used to set or confirm the piping temperatures design basis since the mass and energy releases provided for Case 3L is very conservative. The mass and energy releases 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 mass and energy releases. The maximum allowed (per Technical Specifications) RWST water temperature has been set to 65°F following the issuance of these mass and energy releases. Design experience has shown that the limiting type of accident sequence for maximum containment sump water temperature is a DEPS (Case 6L and 7L) not a DEHL like case 3L. The excess margin in Case 3L's mass and energy releases caused the departure from this trend for the results summarized in this table. Engineering judgement was applied such that the mass and energy releases for Case 3L were not revised and re-issued to eliminate this anomaly.

Table 4.1-21
MAAP5 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	100.6	DG	31.3	28.4	---
1L-DEHL MIN SI	100.6	LHSI	27.4	25.3	---
2L-DEHL MAX SI	100.6	None	16.1	16.8	---
3L-DEHL MIN SI (1)	100.6	DG	19.7	20.8	20.9
4L-DEHL MAX SI	100.6	None	16.9	18.0	33.7
6L-DEPS MIN SI	100.6	DG	21.1	19.7	23.7
7L-DEPS MAX SI	100.6	CIB	20.4	20.9	38.5

Single Failures – Failed Equipment

CIB One train each, QSS, RSS

DG One train each, SI, QSS, RSS

LHSI One LHSI train

(1) As discussed in note (2) in Table 4.1-20, the results for this case should be disregarded relative to setting or confirming the containment design basis due to the excessive conservatisms in the mass and energy releases supplied for Case 3L.

Table 4.1-22
MAAP5 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	22.4
2L-DEPS MAX SI	100.6	CIB	22.7
2L1-DEPS MAX SI	100.6	SW	22.9
2L2-DEPS MAX SI	100.6	RELAY	22.7
3L-DEHL (break into node #5)	100.6	None	23.7

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

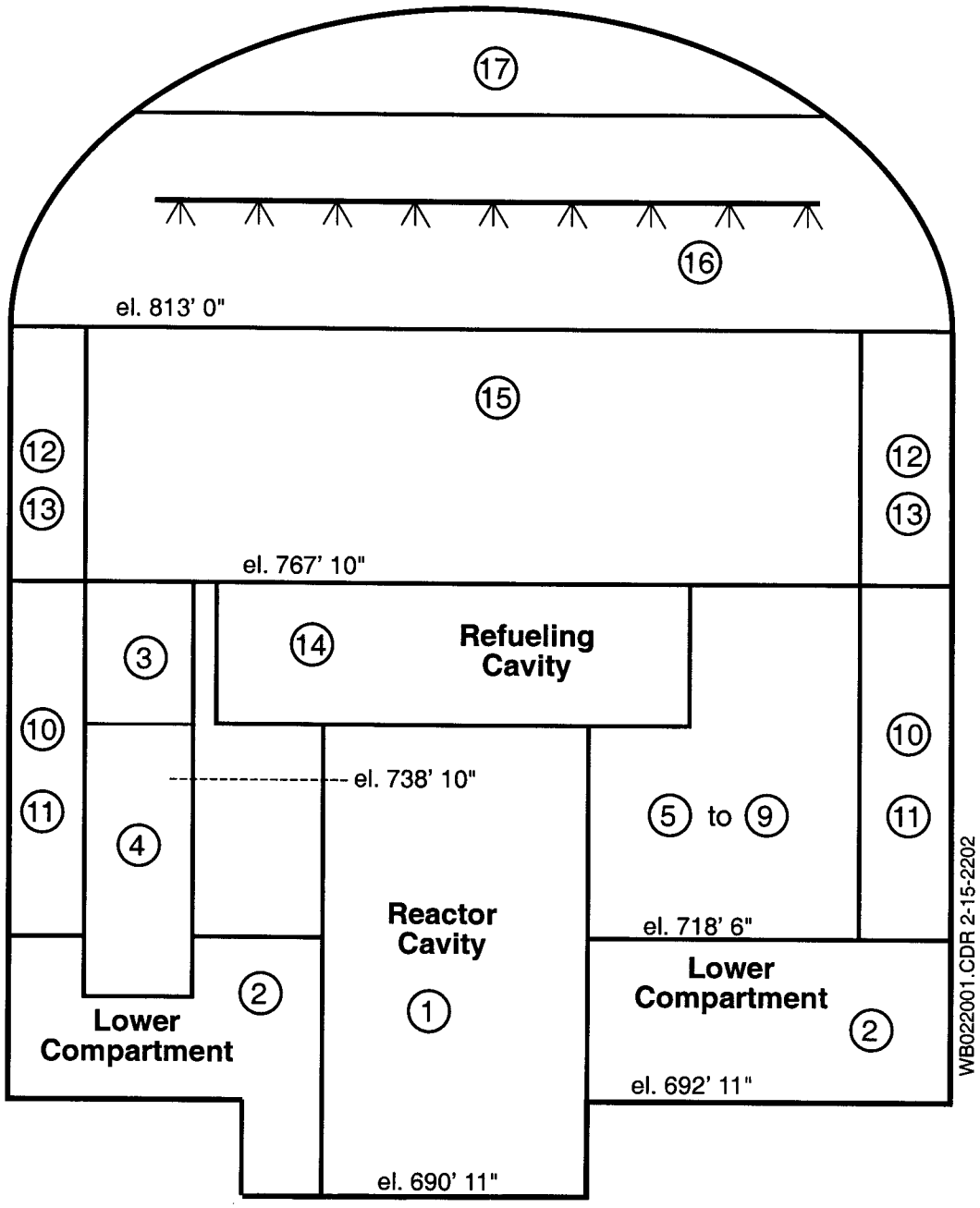


Figure 4.1-1
MAAP5 Containment Nodalization for BVPS-1 and BVPS-2

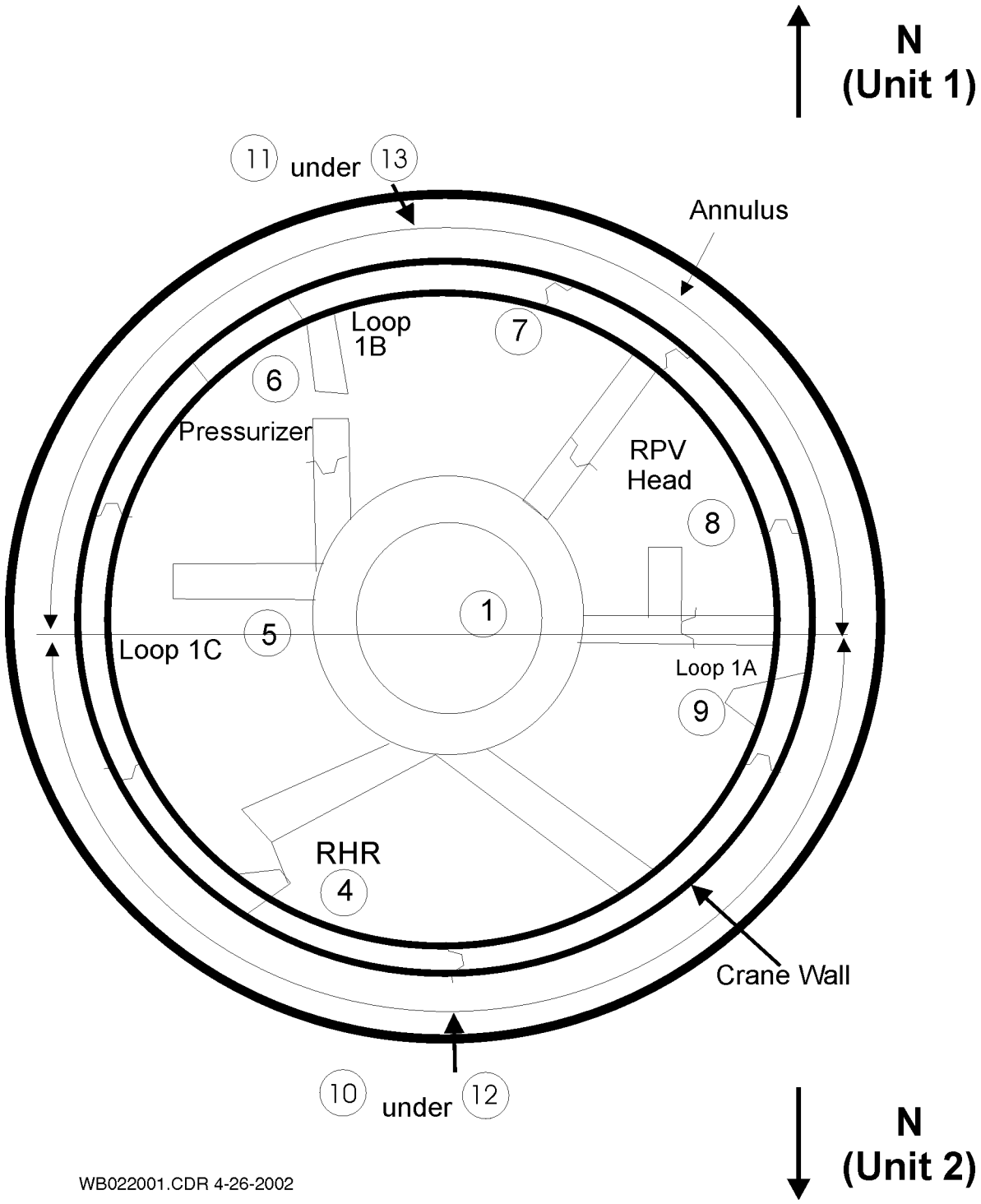


Figure 4.1-2
MAAP5 Containment Nodalization (Plan View) for BVPS-1 and BVPS-2

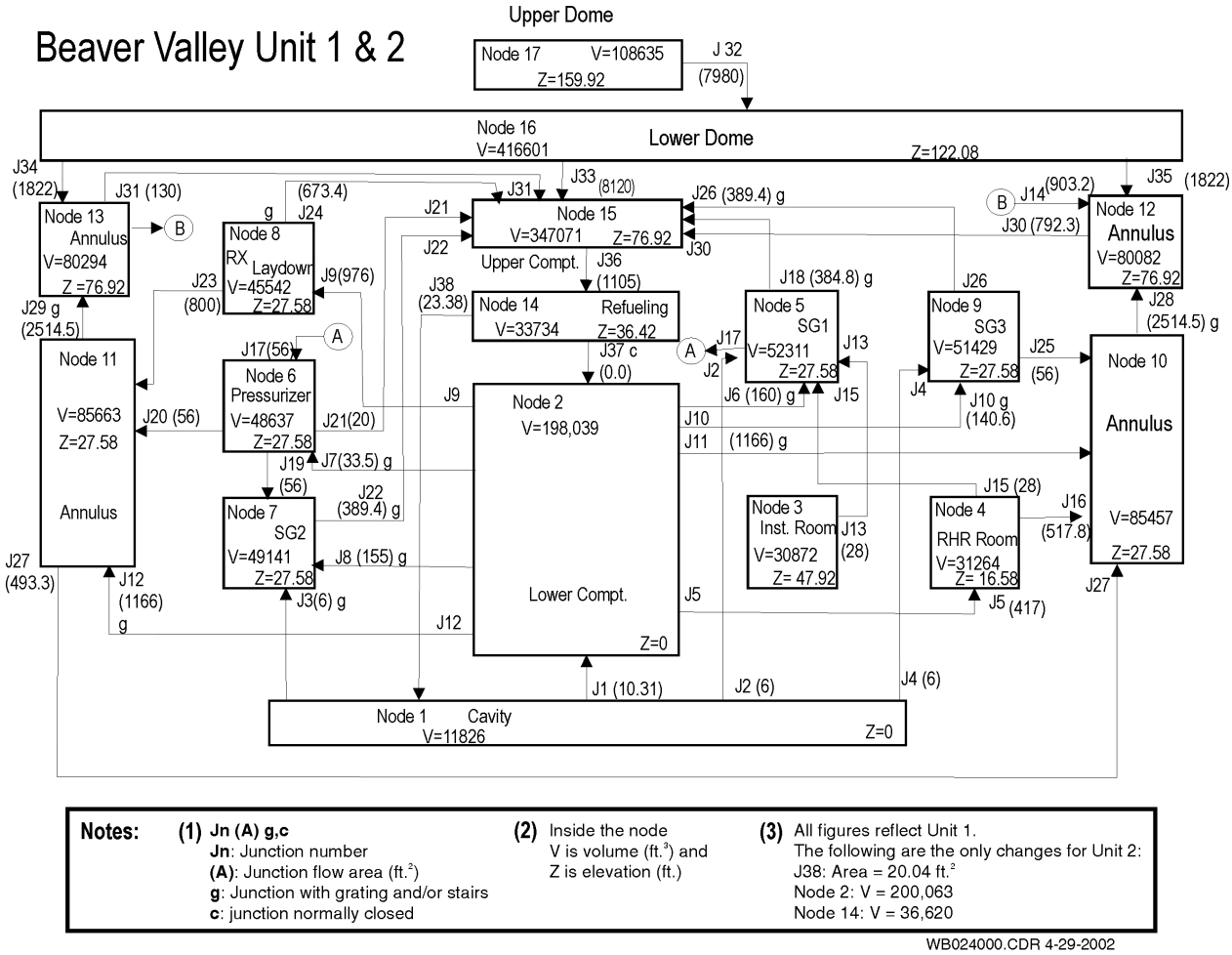


Figure 4.1-3
MAAP5 Node and Junction Arrangement for BVPS-1 and BVPS-2

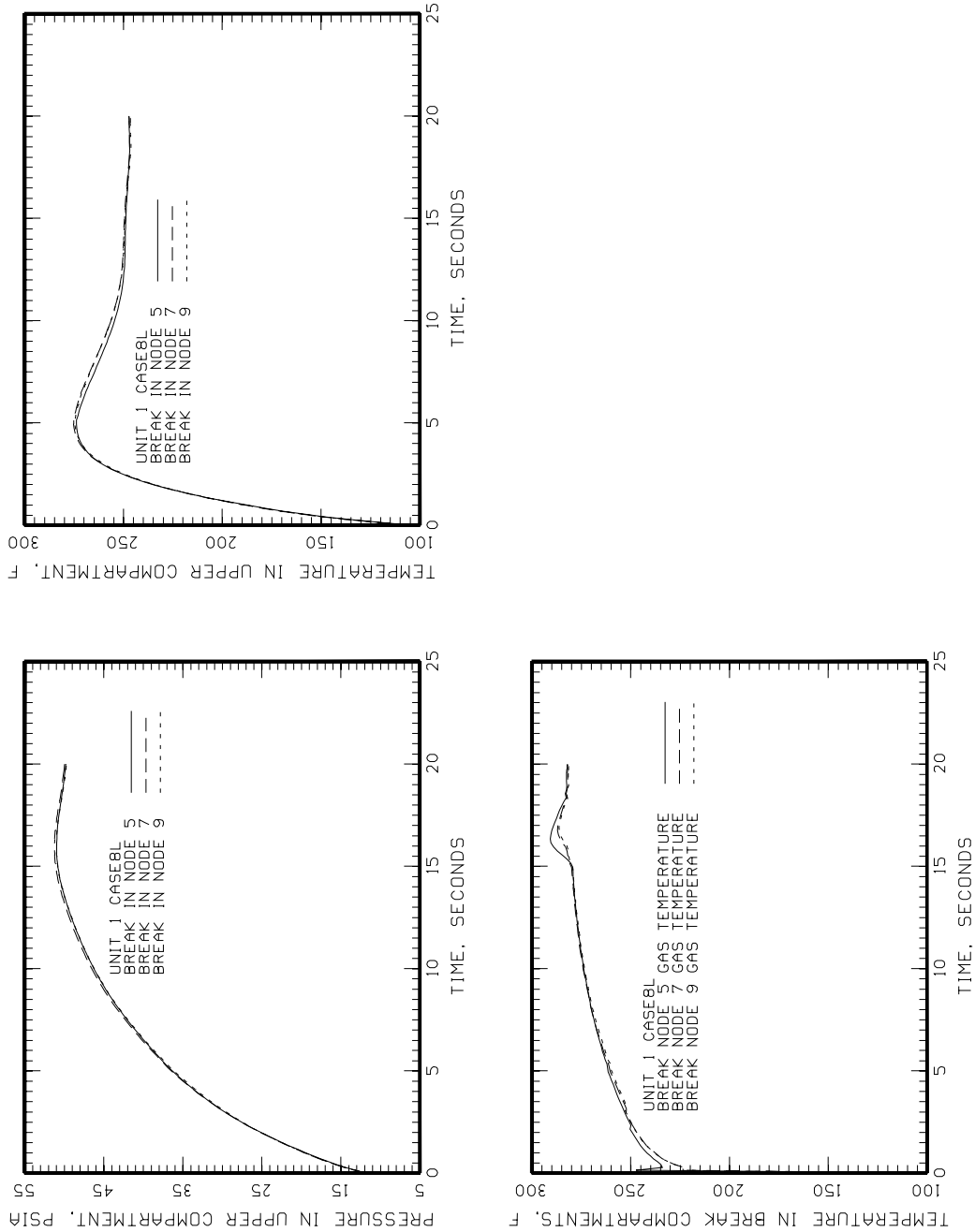


Figure 4.1-4
Composite Figure of the Containment Pressurization and Gas Atmosphere Temperature Histories for the DEHL Analyses in the Three Different Loop Compartments for BVPS-1

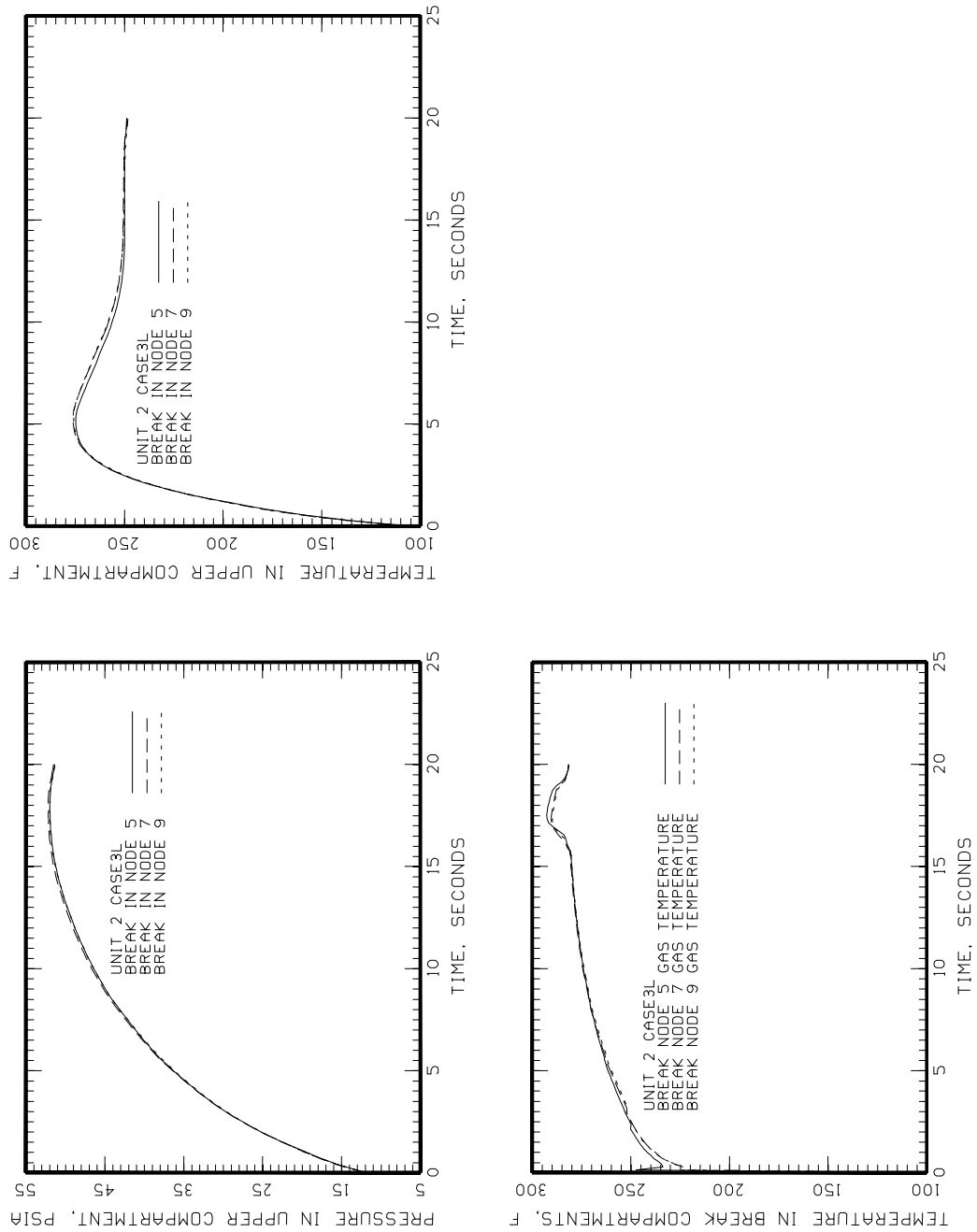


Figure 4.1-5
Composite Figure of the Containment Pressurization and Gas Atmosphere Temperature Histories for the DEHL Analyses in the Three Different Loop Compartments for BVPS-2

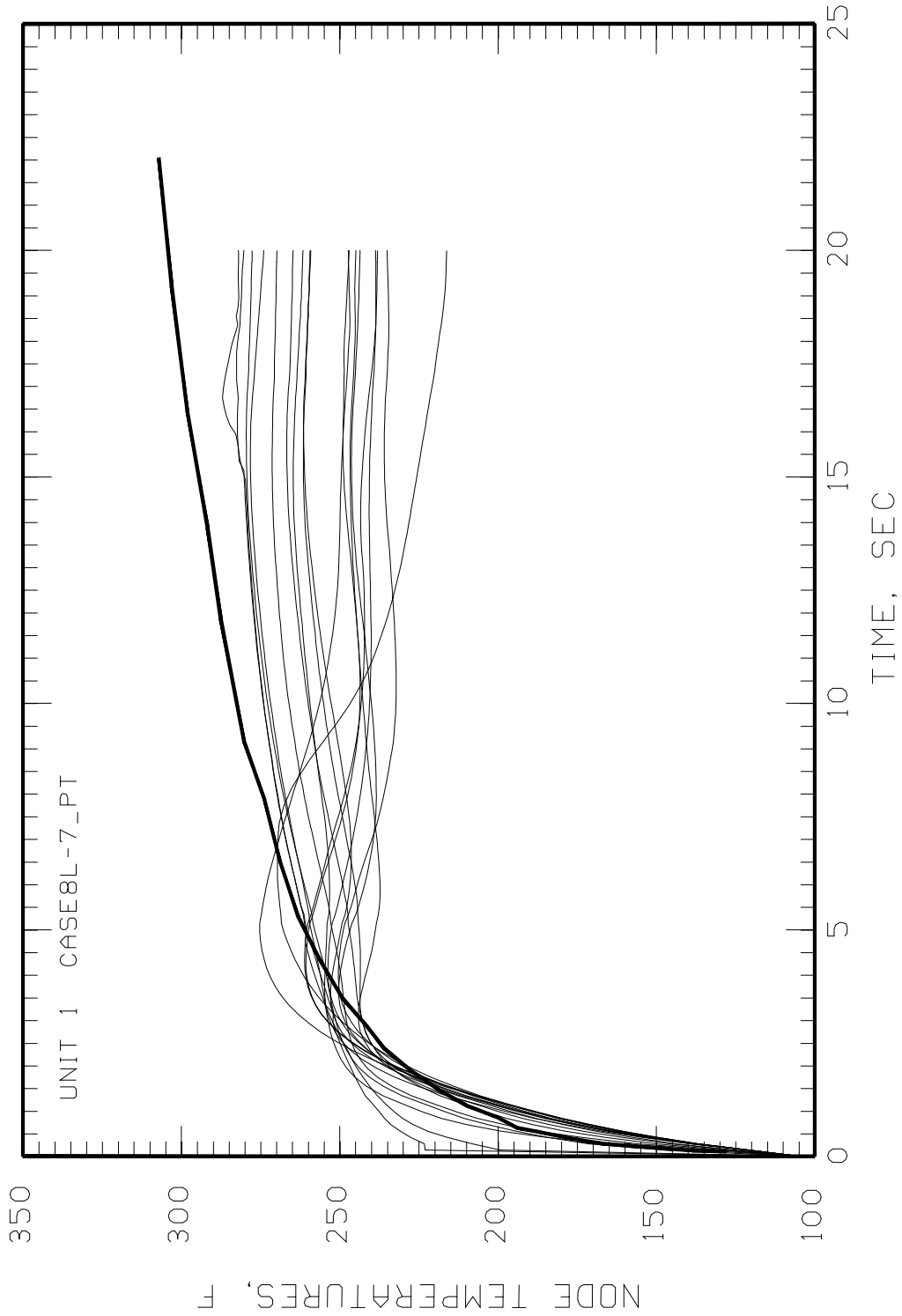
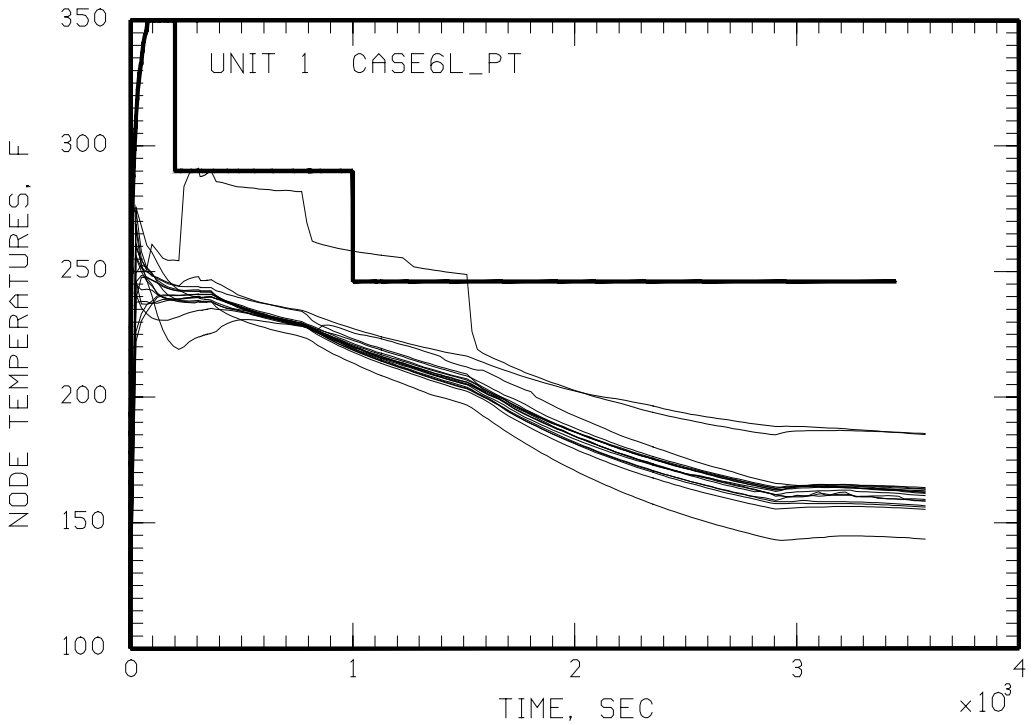


Figure 4.1.6
Comparison of the 17 Nodal Temperatures with the Current EQ Profile for Case 8L (Double-Ended Hot Leg Break) for BVPS-1



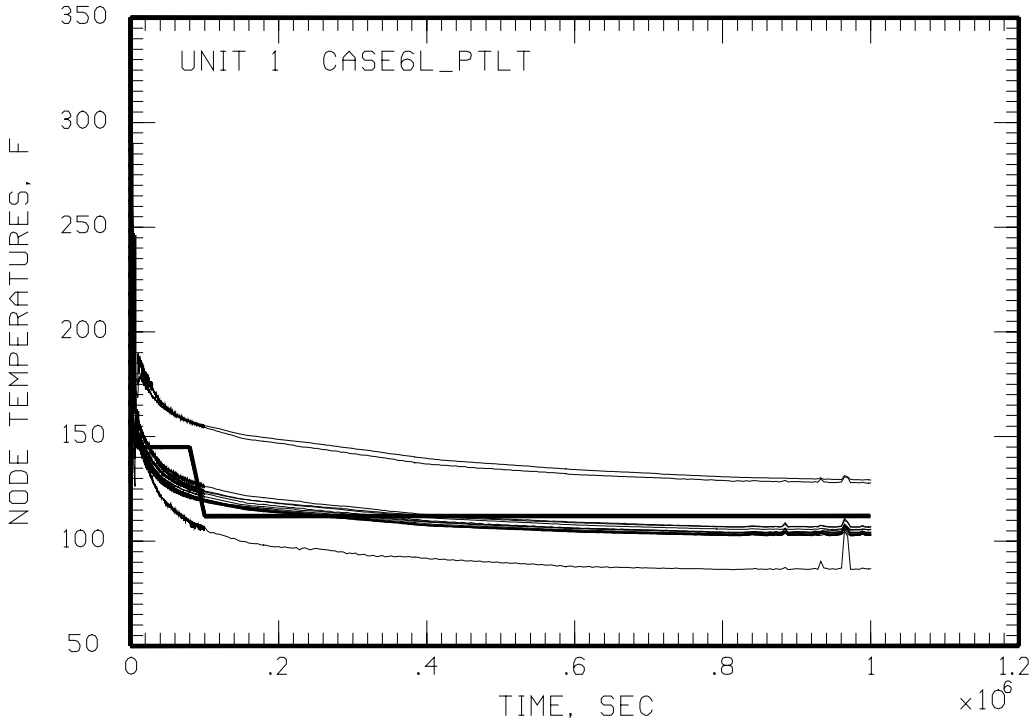


Figure 4.1-7
Comparison of the 17 Nodal Temperatures with the Current EQ Profile for Case 6L (Double-Ended Pump Suction Break) for BVPS-1

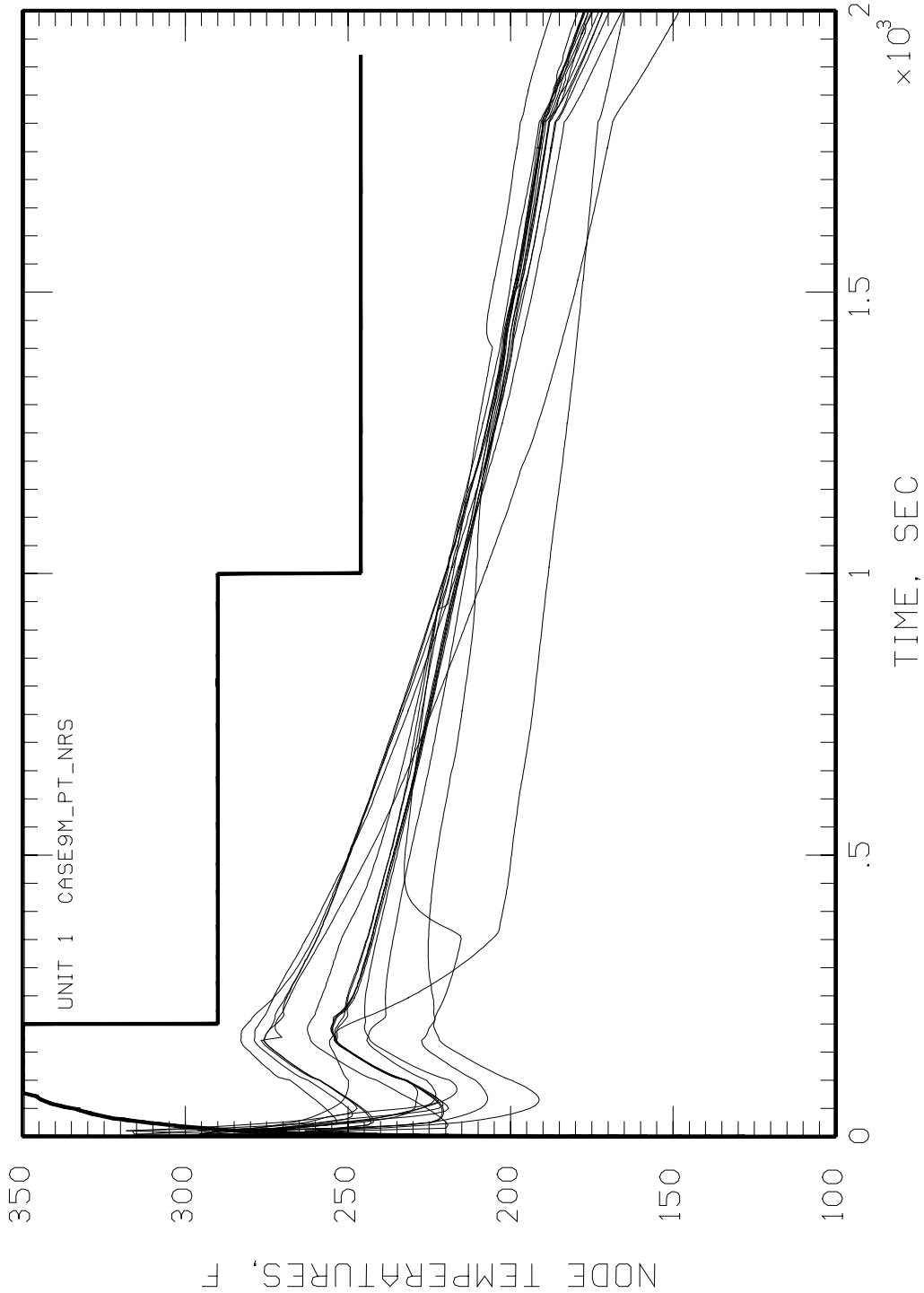


Figure 4.1-8
Comparison of the 17 Nodal Temperatures with the Current EQ Profile for Case 9M for BVPS-1

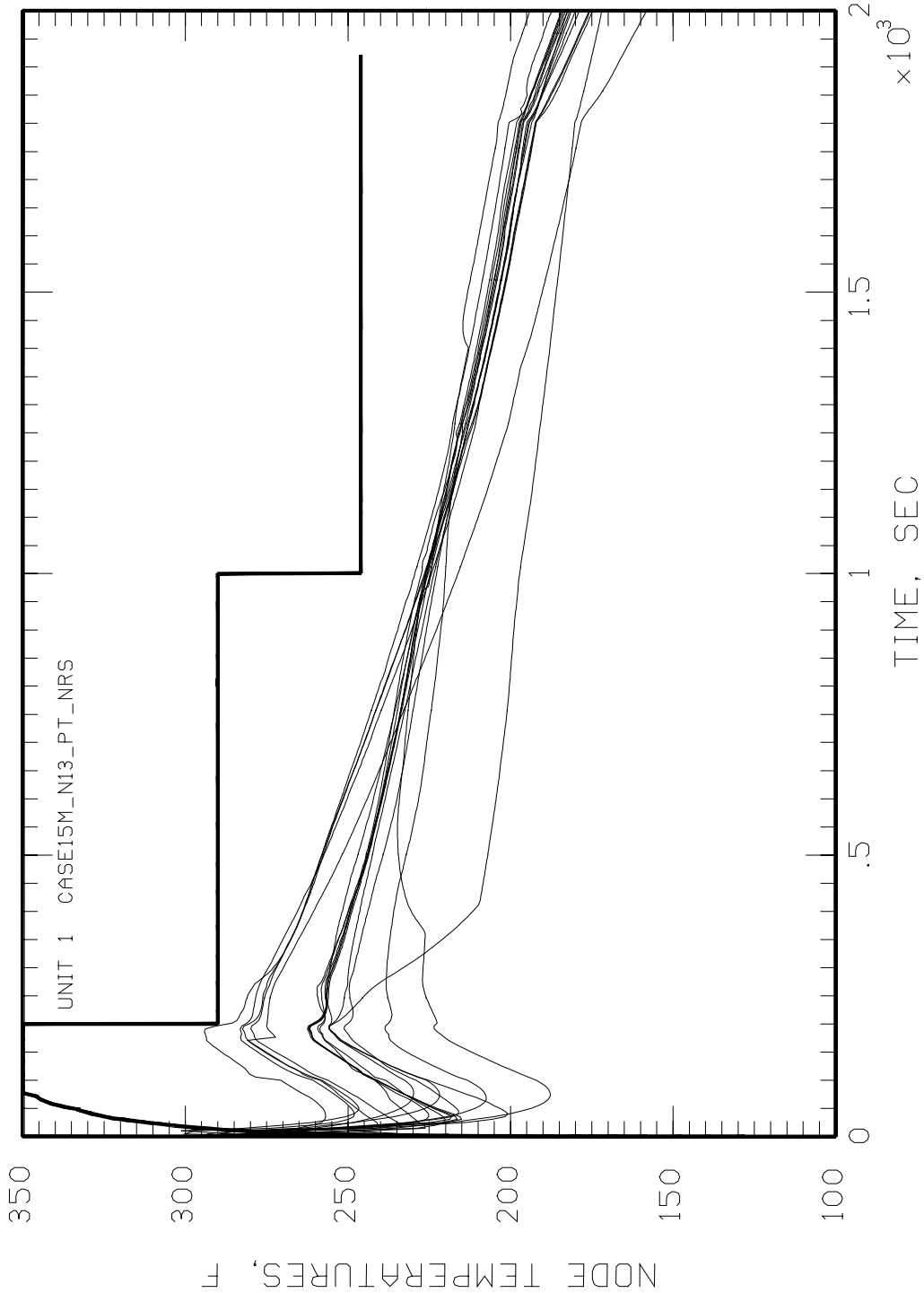


Figure 4.1-9
Comparison of the 17 Nodal Temperatures with the Current EQ Profile for Case 15M for BVPS-1

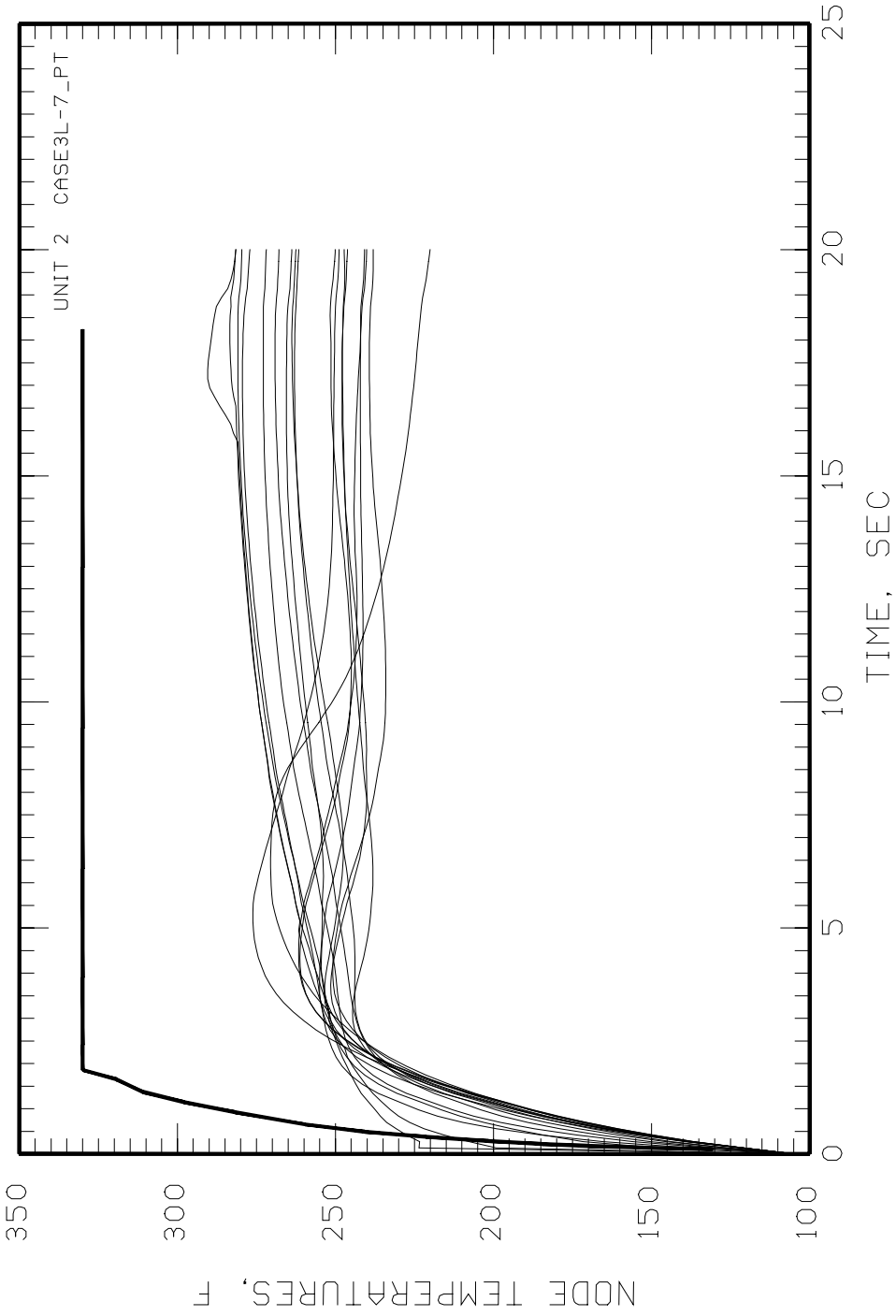


Figure 4.1-10
Comparison of the 17 Nodal Temperatures with the Current EQ Profile for Case 3L (Double-Ended Hot Leg Break) for BVPS-2

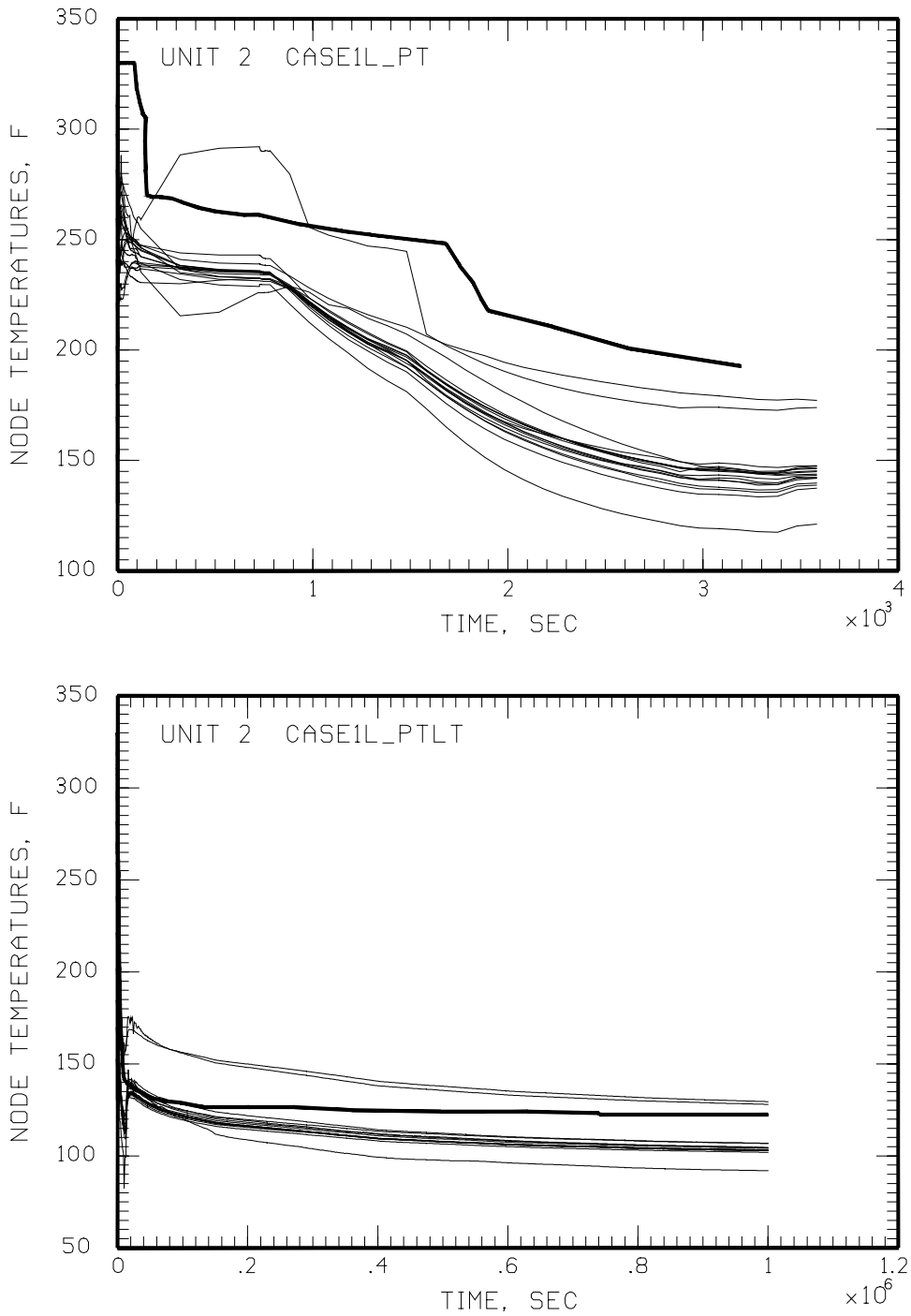


Figure 4.1-11
Comparison of the 17 Nodal Temperatures with the Current EQ Profile for Case 2L (Double-Ended Pump Suction Break) for BVPS-2

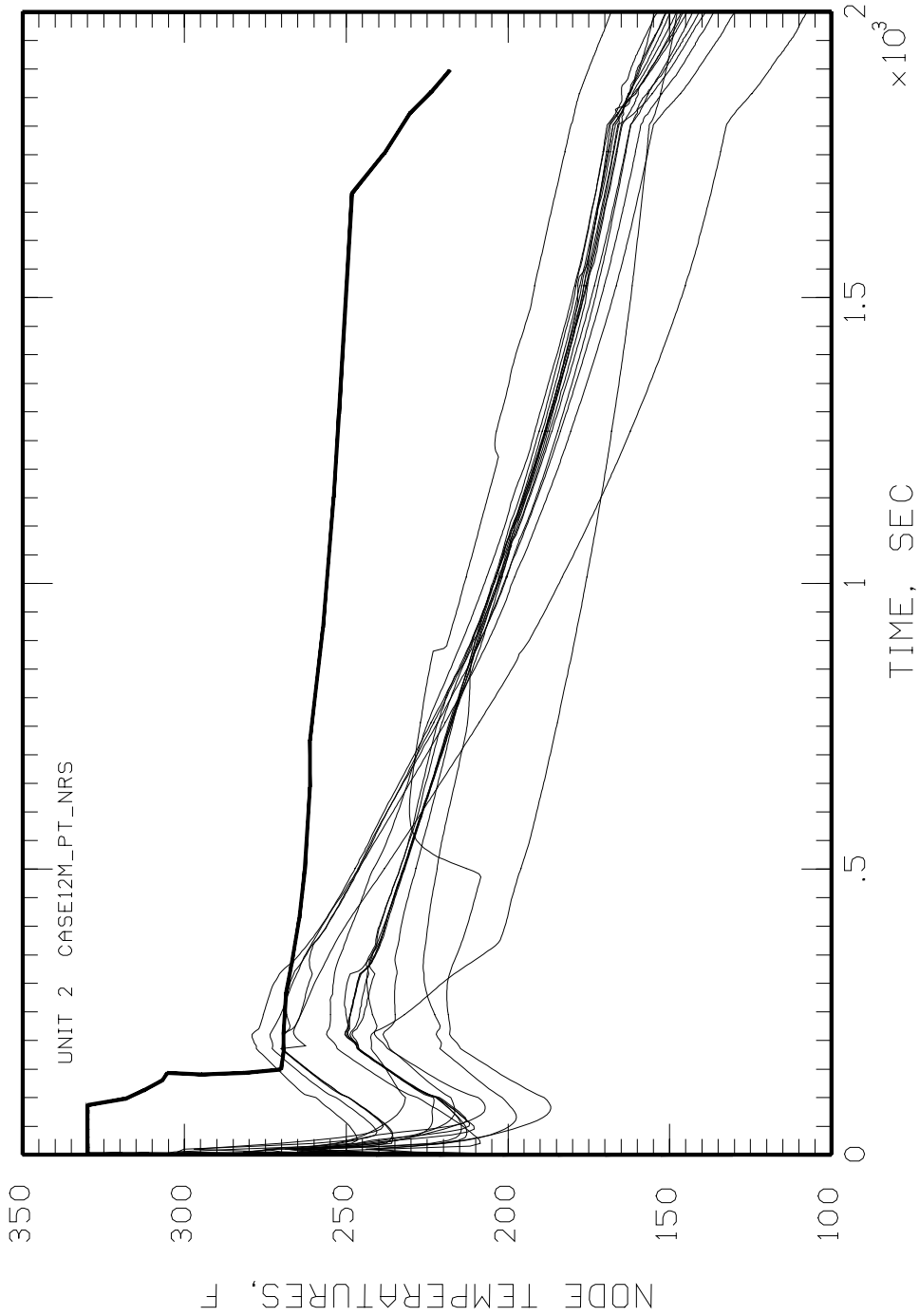


Figure 4.1-12
Comparison of the 17 Nodal Temperatures with the Current EQ Profile for Case 12M for BVPS-2

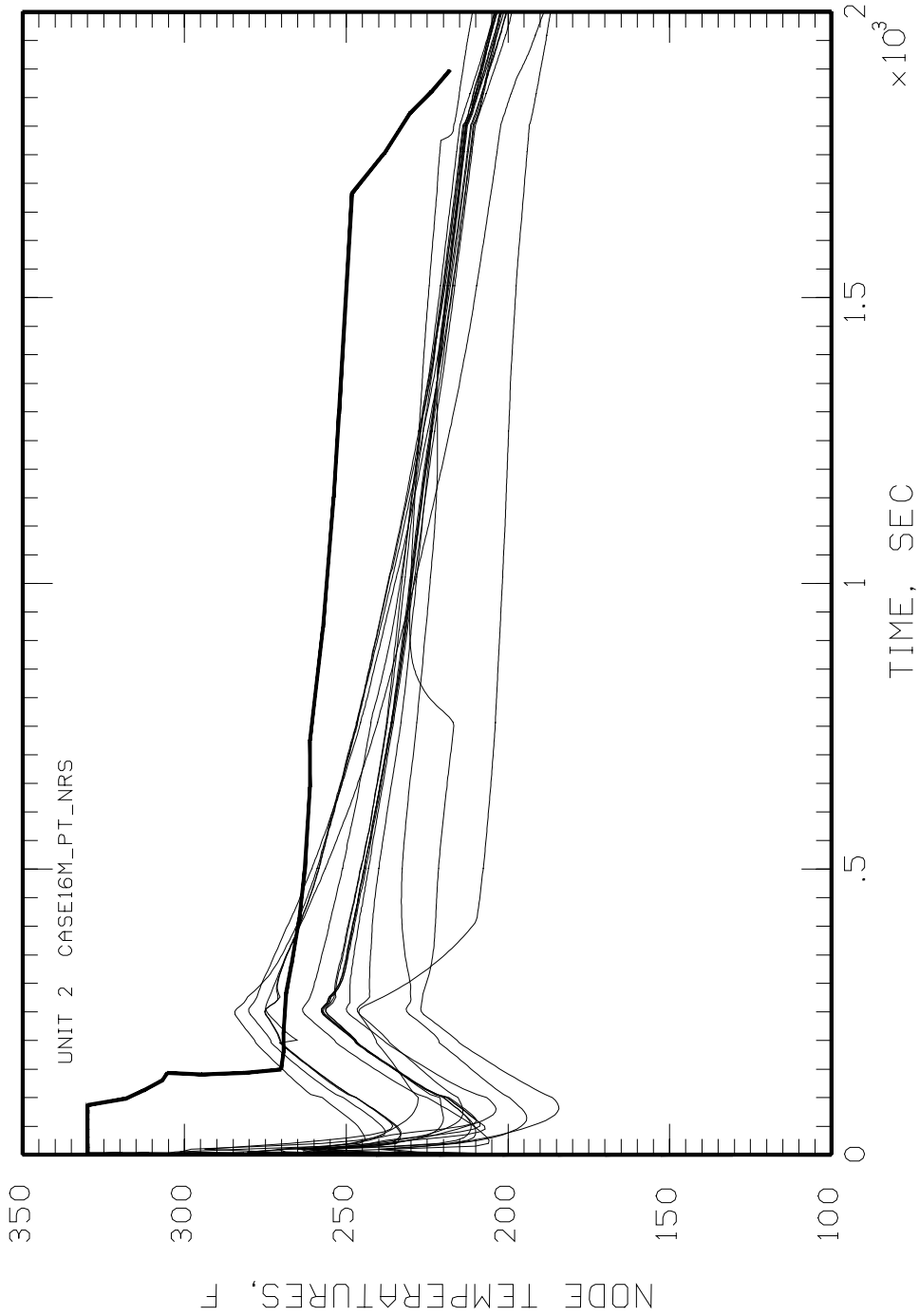


Figure 4.1-13
Comparison of the 17 Nodal Temperatures with the Current EQ Profile for Case 16M for BVPS-2

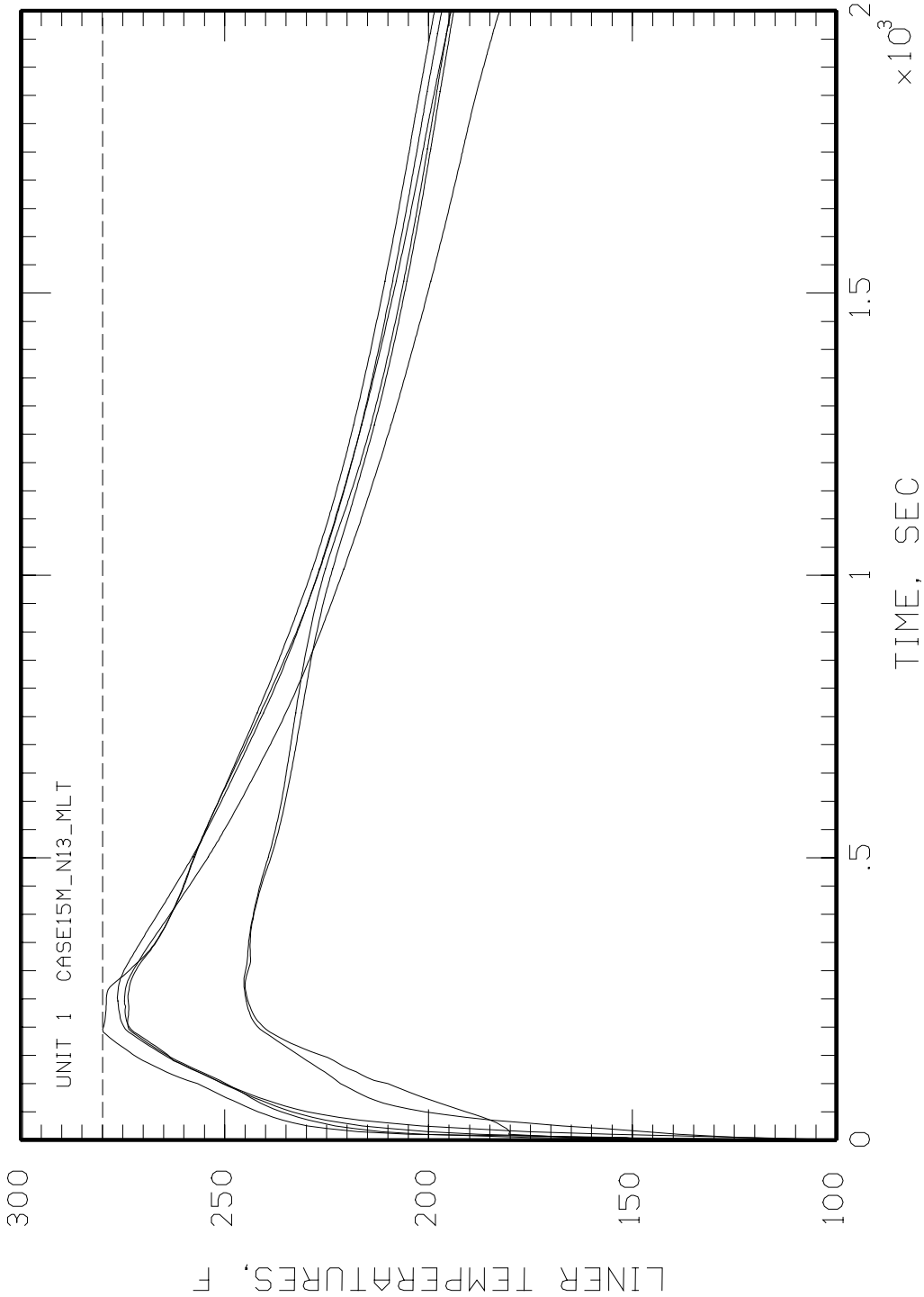


Figure 4.1-14
Comparison of Liner Temperatures with their Temperature Limit for Case 15M (Break in Annular Compartment) for BVPS-1

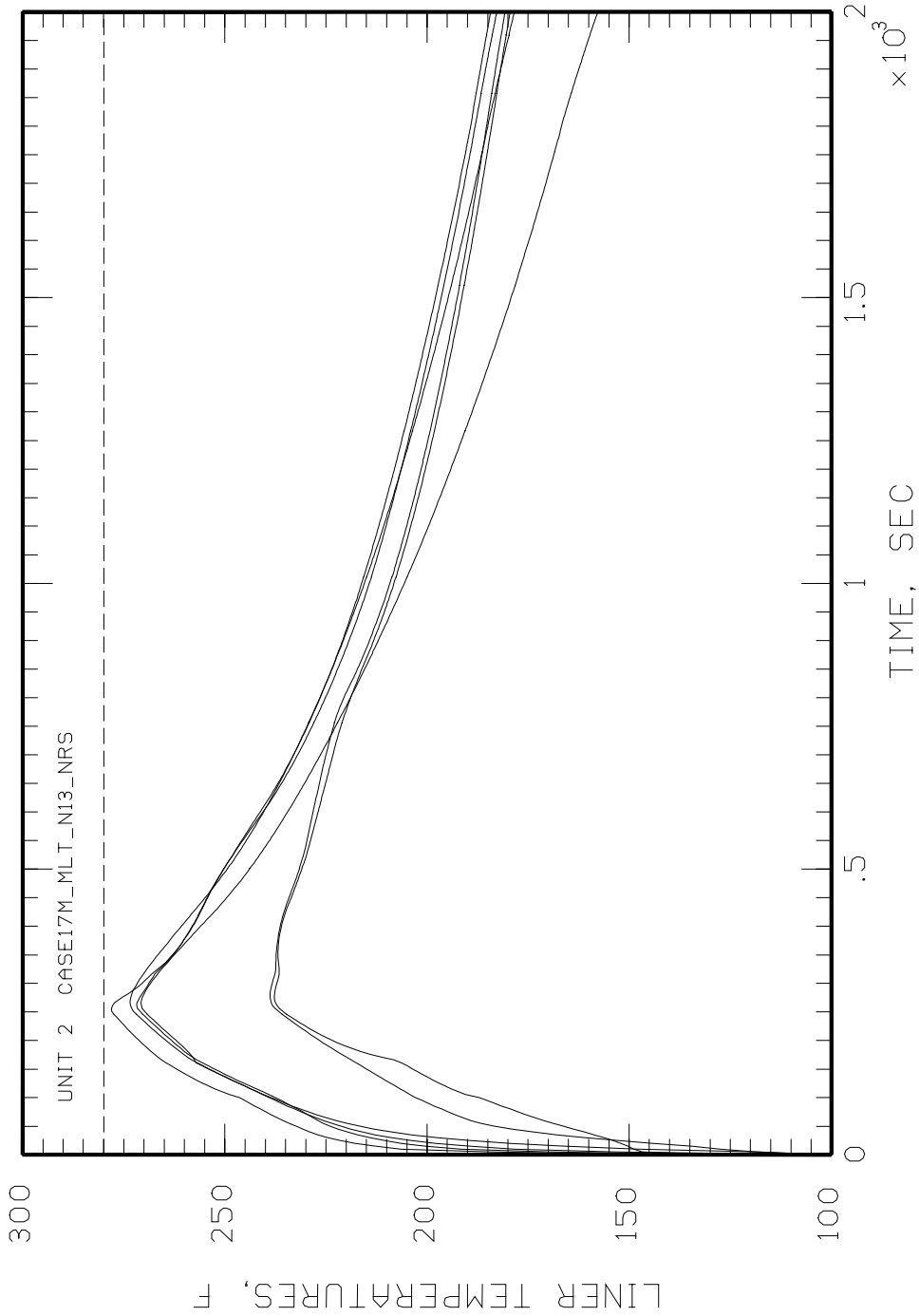


Figure 4.1-15
Comparison of Liner Temperatures with their Temperature Limit for Case 17M (Break in Annular Compartment) for BVPS-2

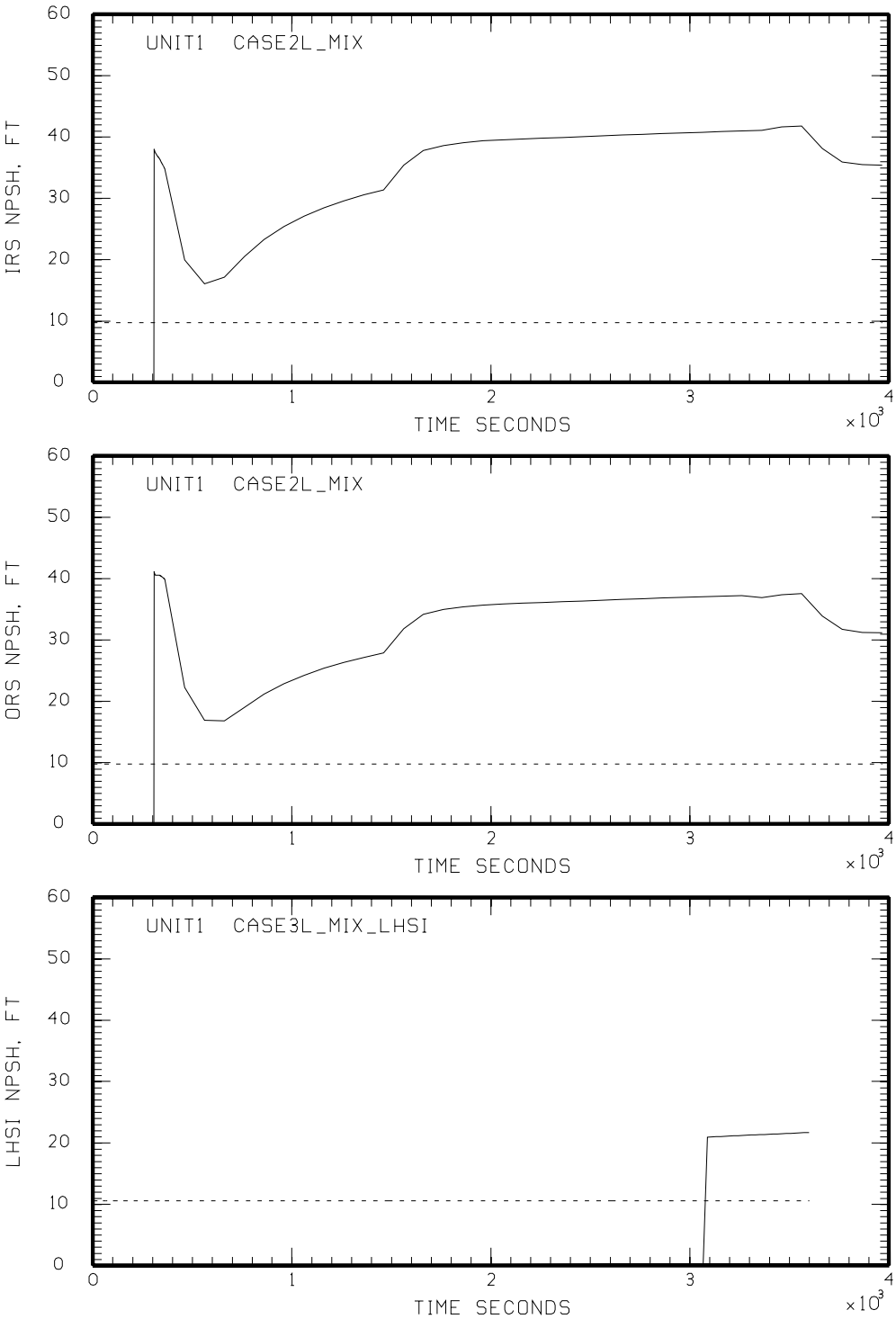


Figure 4.1-16
Available NPSH for BVPS-1 RS and LHSI Pumps for Case 6L

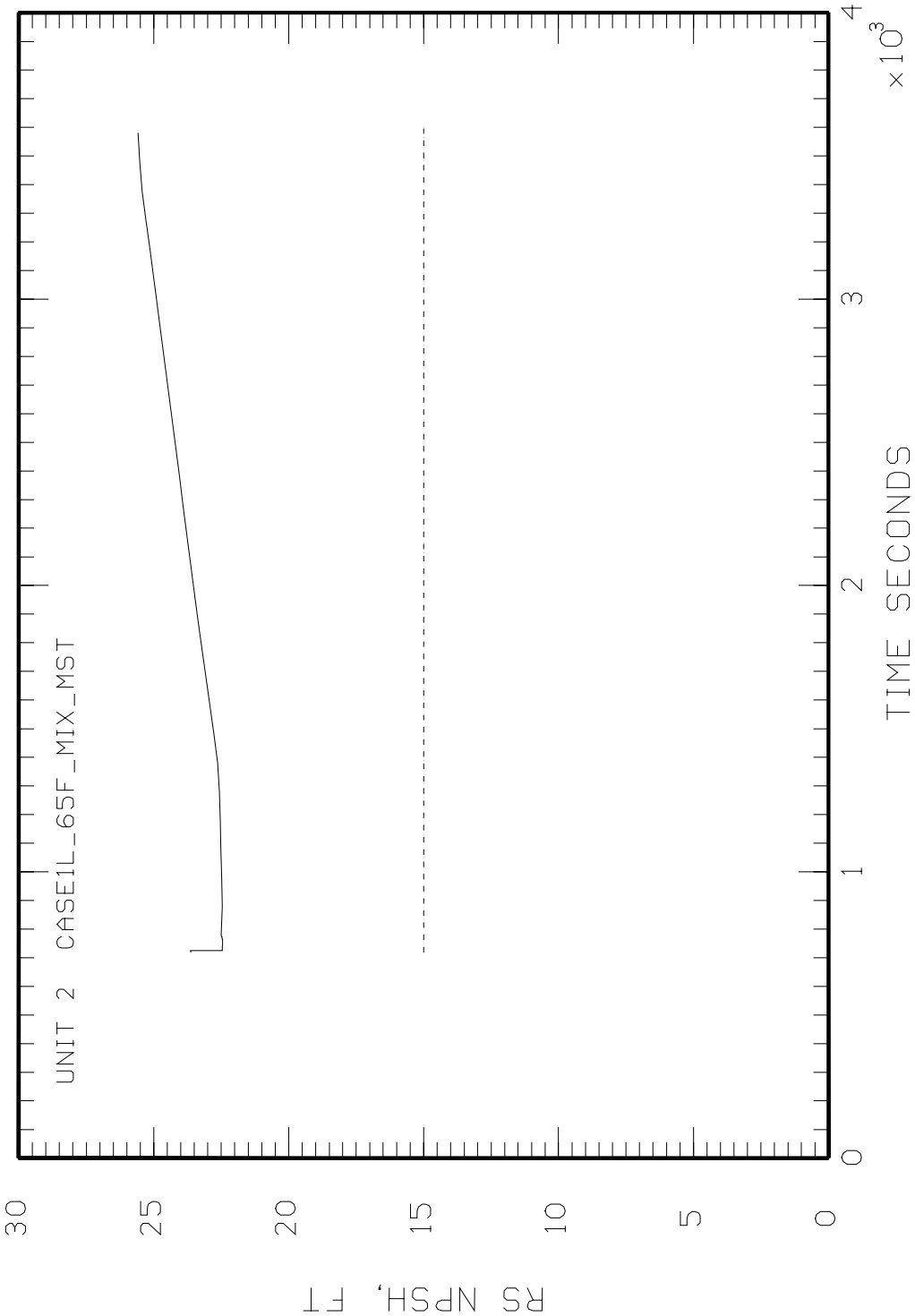


Figure 4.1-17
Available NPSH for BVPS-2 RS Pump for Case 1L

5.0 RADIOLOGICAL ASSESSMENT

5.1 Introduction

This chapter addresses the potential radiological impacts of changing the normal operating pressure in the BVPS-1 and BVPS-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".

It is noted that several changes are being included with the atmospheric conversion modification. A complete list of the proposed design changes is provided in Section 1.1.2. Only those design changes that impact radiological consequences will be addressed in this section.

As holder of operating licenses issued prior to January 10, 1997, and in accordance with 10CFR50.67 (Reference 1) and Standard Review Plan 15.0.1 (Reference 2), FENOC proposes to voluntarily revise the accident source term used in the BVPS-1 and BVPS-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).

A selective implementation is a modification of the plant design basis that (1) is based on one or more of the five characteristics of an AST and/or (2) reevaluates a limited subset of the design basis radiological analyses. The BVPS implementation is based on all five of the characteristics of AST.

- Core inventory and radionuclide composition
- Gap inventory release magnitudes
- Chemical form of the radionuclides
- Physical form of the radionuclides
- Timing of the release of these radionuclides

The first use of the AST for BVPS was a selective application to revise the Fuel Handling Accident (FHA) in order to justify certain changes in plant operation and configuration during fuel movements. Specifically, the reanalysis justifies reducing the minimum decay time before commencing refueling operations from 150 hours to 100 hours. The analysis demonstrated that acceptable doses would occur at both the EAB and in the control room, without taking credit for ventilation systems filtration, or control room isolation. The analysis was reviewed and approved by the NRC in its SER for OL Amendments No. 241 and 121 (Reference 45).

The proposed selective application of the AST for BVPS is to support atmospheric conversion. 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).

The Main Steam Line Break (MSLB) inside containment will also result in a pressure transient inside containment. However, 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. This conclusion remains valid with atmospheric conversion, including all of the design changes being incorporated in connection with it.

Consequently, in support of atmospheric conversion, the dose consequences at the control room, and site boundary are only evaluated for the LOCA and the CREA. The methodology and scenarios discussed in the BVPS-1 and BVPS-2 UFSARs for the LOCA and CREA site boundary and control room dose analyses are being updated to reflect the guidance provided in Regulatory Guide 1.183 (Reference 3).

Within the foreseeable future, FENOC intends to submit a LAR to operate at an uprated power level. To this end, the LOCA and the CREA analyses supporting atmospheric conversion are conservatively based on an equilibrium core and NSSS power level (2910 MWT) that is representative of uprated conditions; and reflect bounding parameter values to encompass an event at either unit.

In addition, the updated LOCA and CREA dose analyses reflect the following updated inputs, as well as the “other” changes included as part of the atmospheric conversion package.

Changes

1. The analyses reflect the change in the RWST low level transfer setpoints which impacts the time the Emergency Core Cooling System (ECCS) switchover from the injection to the recirculation mode as this change could potentially impact the timing of the start of iodine releases due to post-LOCA RWST back leakage.

2. The analyses reflect the change in the RWST temperature range as this change impacts the iodine release fraction via the RWST vent due to post-LOCA RWST back leakage.

3. The analyses take no credit for the control room emergency bottled air pressurization system (CREBAPS) for accident mitigation functions. This design feature is no longer considered necessary since the BVPS containments will no longer be required to return to sub-atmospheric conditions within sixty minutes following a LOCA.

4. The analyses address the elimination of the 60 minute time delay upon receipt of the CIB signal which is currently used to delay the start of the control room emergency ventilation system (CREVS) pressurization fans to accommodate control room pressurization after one hour.

5. The analyses credits the revised Technical Specification acceptance criteria for the BVPS-1 control room HEPA and charcoal filters which will make the BVPS-1 acceptance criteria similar to the more limiting criteria currently listed for the BVPS-2 control room filters.

6. The analyses takes no credit for the HEPA and charcoal filters in the Supplementary Leak Collection and Release System (SLCRS) exhaust flow for accident mitigation functions.

7. The automatic initiation feature of the safety related control room radiation monitors to initiate the control room emergency pressurization system is not credited. Note that this design feature is currently not credited in the LOCA and CREA analyses, and therefore has no impact on the dose consequences of the LOCA and the CREA. However, as discussed later in this section, the impact on other design basis accident analyses is evaluated.

Updated Inputs

1. The analyses utilize conservative estimates of control room unfiltered inleakage that envelope the results of recent tracer gas testing performed in the year 2001, and provide margin for potential deterioration between surveillance tests.

2. The analyses utilize updated control room atmospheric dispersion factors using ARCON96 methodology.

3. The analyses conservatively reflect manual initiation of CREVS. The control room is assumed to be pressurized via manual initiation of the CREVS by T=30 mins after the LOCA or CREA.

4. As discussed in Section 5.3.6.2, the control room will continue to have three (3) 100% CREVS systems. Two (2) are fully automatic and one (1) is manual. The auto systems start automatically on a CIB. The manual system would only be required in the unlikely event that one (1) auto system is out of service and the second auto system fails to start.

5. The LOCA analysis does not take credit for the HEPA filter in the Emergency Response Facility (ERF) normal operation ventilation intake flow, or the HEPA and charcoal filters in the ERF post accident ventilation recirculation flow.

To ensure a consistent design basis as it relates to radiological analyses impacted by atmospheric conversion, this chapter also includes a summary of the impact of long-term post LOCA

containment leakage resulting from atmospheric conversion and associated use of AST methodology on the:

- post-LOCA operator exposure while performing vital functions in areas impacted by the design change including establishing habitability of the Emergency Response Facility,

- post-LOCA component of the Technical Information Document used to evaluate equipment qualification of Class 1E equipment located in impacted areas

- 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).

Also included is a radiological assessment of the impact of the changes listed above, on control room habitability for the remaining design basis accidents.

Note that with the exception of the BVPS-1 and BVPS-2 Main Steam Line Break (MSLB) and the BVPS-1 Locked Rotor Accident (LRA), the remaining design basis events are not adversely impacted by changes included with atmospheric conversion. The CREBAPS is currently credited for the BVPS-1 and BVPS-2 MSLB and the BVPS-1 LRA. In addition, the automatic CREBAPS initiation function of the control room monitors is credited in the BVPS-1 LRA.

The dose impact of not crediting CREBAPS on control room habitability following a BVPS-1 and BVPS-2 MSLB and a BVPS-1 LRA is assessed using existing licensing basis methodology / assumptions / regulatory acceptance criteria, and is not included in the selective application of AST methodology.

In accordance with current licensing basis, the control room will be assumed to be pressurized at T=30 mins following a MSLB at either unit. However, unlike the current design basis, credit will be taken for manual initiation of CREVS, instead of CREBAPS followed by CREVS.

The BVPS-1 LRA continues to depend on the control room safety-related radiation monitors to alarm on high radiation levels in the control room. However, instead of automatic initiation of CREBAPS subsequently followed by CREVS, the updated analysis reflects control room pressurization at T=1 hr. Pressurization begins on manual initiation of CREVS upon acknowledgement of a high radiation signal from the radiation monitors.

5.2 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 Accident Assessment

The changes proposed by this application will potentially impact the reported dose consequences of the BVPS-1 and BVPS-2 LOCA, CREA, MSLB and the BVPS-1 LRA. The radiological consequences of the remaining design basis accidents are not adversely impacted by this application.

Summarized below are the changes / updated inputs proposed by this application, and the corresponding design basis analyses impacted by the change.

- Atmospheric conversion will only impact the site boundary and control room doses following a LOCA and the CREA. ERF habitability following a LOCA will also be impacted. Long term post LOCA containment leakage resulting from atmospheric conversion may also impact the post LOCA component to the Total Integrated EQ dose, as well as the estimated operator exposure while performing vital functions, in areas affected by the design change. Long term post LOCA containment leakage will also impact iodine loading on post accident charcoal filters.

- The change in the RWST low level transfer points and temperature range can affect the site boundary doses, and control room or ERF habitability following a LOCA.

- Elimination of the accident mitigation function of the HEPA and charcoal filters in the SLCRS ventilation exhaust system can affect the site boundary doses, and control room or ERF habitability following a LOCA.

- Update of the Technical Specification acceptance criteria for the BVPS-1 control room HEPA and charcoal filters to criteria similar to that currently listed for the BVPS-2 control room filters will render the existing dose consequences for BVPS-1 accidents conservative as the BVPS-2 filter acceptance criteria is more limiting.

- Elimination of the accident mitigation capability of the CREBAPS and the sixty minute time delay associated with the CREVS will only impact control room habitability following a LOCA (both units), the MSLB (both Units), and the LRA (BVPS-1).

- Elimination of credit for the automatic initiation function of the safety related control room radiation monitors to initiate the control room emergency pressurization system will only impact control room habitability following a BVPS-1 LRA.

- Use of conservative estimates of control room unfiltered inleakage that envelope the results of recent tracer gas testing performed in the year 2001, and provide margin for potential deterioration between surveillance tests for the LOCA and the CREA do not invalidate the inleakage assumptions used in the remaining design basis events. The results of the tracer gas tests and leakage measurements provide assurance that the values for unfiltered inleakage into the control room used in the existing analyses remains valid at this time.

- Use of ARCON 96 methodology in development of control room atmospheric dispersion factors for the LOCA and the CREA does not invalidate the control room atmospheric dispersion factors utilized for the other design basis accidents using existing licensing basis methodology.

- The assumption of manual initiation of CREVS will only impact control room habitability following a LOCA (both Units), CREA (both Units), MSLB (both units) and the BVPS-1 LRA. The remaining accidents do not take credit for the control room emergency ventilation system.

- Not taking credit for the accident mitigation function of the HEPA filter in the ERF normal operation intake ventilation flow, and the HEPA and charcoal filter in the post accident ERF recirculation flow is conservative for establishing impact on ERF habitability following a LOCA.

5.3.1 Regulatory Approach

In 1962, the U.S. Atomic Energy Commission (AEC) published the technical information document TID-14844, “Calculation of Distance Factors for Power and Test Reactors” (Reference 7), which specified a release fraction of the fission products from the core to the reactor containment, or “source term”, to be considered in the event of a postulated design basis accident (DBA) involving a “substantial meltdown of the core” in a Light Water Reactor. These

fractions were based on small-scale experiments performed in the late 1950s in which irradiated UO_2 pellets were heated to accident temperatures.

The TID-14844 accident source term formed the basis for NRC Regulatory Guide 1.4 (Reference 46), which was used to determine compliance with the NRC reactor siting criteria, 10 CFR Part 100.11 (Reference 6). It was also used to evaluate other plant performance requirements including control room habitability requirements given in 10CFR50 Appendix A, GDC-19 (Reference 47) and expanded in NUREG 0800, Standard Review Plan 6.4 (Reference 26), of 5 rem whole body, 30 rem beta and 30 rem thyroid. The TID-14844 source term was also used for evaluation of equipment qualification (EQ) of class 1E electrical components in accordance with 10CFR50.49 (Reference 48), and evaluation of vital area accessibility post-LOCA in accordance with NUREG-0737 (Reference 8).

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 49), which used this research to provide estimates of an accident source term that was more physically based and that could be applied to Light Water Reactors. This effort crystallized into regulatory acceptance for an alternative source term and the issuance of 10CFR50.67 and Regulatory Guide 1.183, which formalized the applicability of AST for operating reactors.

With this application, BVPS 1 and 2 propose a "selective" implementation of the AST as defined in RG 1.183, Section 1.2.2 and SRP 15.0.1, Section III.2.b.2 to implement atmospheric conversion.

Consequently, as noted in the referenced guidance, upon receipt of NRC acceptance of the proposed change, the design basis for the site boundary and control room radiological assessment of the LOCA and the CREA will be 10CFR50.67, as supplemented by Regulatory Guide 1.183. Note that as part of this application the design basis methodology utilized for the development of the on-site atmospheric dispersion factors for the CREA and the LOCA is changed to ARCON96.

The design basis for the radiological assessment of the remaining accidents (with the exception of the FHA, which is already based on 10CFR50.67), will continue to be 10CFR100.11 (Reference 6) and as documented in the current UFSAR. Note that the impact of eliminating automatic initiation of CREBAPS via the radiation monitors, as well as the mitigation function of CREBAPS, on control room doses following a BVPS-1 and BVPS-2 MSLB and BVPS-1 LRA has been assessed using existing licensing basis methodology / assumptions, and is not included in the selective application of AST methodology.

The acceptance criteria for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses for the LOCA and the CREA are based on 10CFR Part 50 § 50.67 and Section 4.4 Table 6 of Regulatory Guide 1.183 (also noted in Table 1 of SRP 15.0.1):

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, should not receive a radiation dose in excess of the accident specific total effective dose equivalent (TEDE) value noted in Reference 3, Table 6.

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), should not receive a radiation dose in excess of the accident specific TEDE value noted in Reference 3, Table 6.

The acceptance criteria for the Control Room Dose for the LOCA and the CREA are based on 10CFR Part 50 § 50.67:

> Adequate radiation protection is provided to permit occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

As required by Regulatory Guide 1.183, to ensure consistency in design basis as related to the analyses impacted by the design change, an assessment is made of the impact of atmospheric conversion and use of AST methodology on associated equipment qualification and post LOCA vital access doses including habitability of the Emergency Response Facility (ERF). Note that the changes included with atmospheric conversion have no impact on radiological equipment qualification and Vital Access doses.

- Equipment Qualification: The equipment qualification (EQ) assessment takes into consideration the impact of long-term post-LOCA airborne radiation levels in the plant, and the increased buildup of activity on the Supplementary Leak Collection and Release System (SLCRS) and control room HEPA/charcoal filters, due to atmospheric conversion and resulting 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 uprated 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 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. The NRC staff is assessing the effect of increased cesium releases (associated with AST methodology) on EQ doses as a generic safety issue (GSI). Pending resolution of this issue, licensees may use either the AST or TID 14844 assumptions for performing the required EQ analyses.

- Vital Area Access: The vital access assessment takes into consideration the impact, if any, 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 44) 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 AST (Reference 53), 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. Airborne levels in the plant due to long term containment leakage resulting from atmospheric conversion were not addressed.

  The BVPS licensing basis for vital access is based NUREG-0737 II.B.2 and only considers "contained" sources. This approach is documented 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; and 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).

  Habitability of the BVPS ERF (which houses both the Technical Support Center (TSC) and the Emergency Operations Facility (EOF) is assessed using a methodology similar to that utilized for the control room (i.e., Regulatory Guide 1.183). The assessment demonstrates compliance with paragraph IV.E.8 of Appendix E, to 10 CFR Part 50, Emergency Planning and Preparedness for Production and Utilization Facilities, regulatory guidance provided in

Supplement 1 of NUREG-0737 (Reference 12) and current BVPS licensing basis, without taking credit for operation of ventilation equipment in the ERF.

Also included is an assessment of the impact of increased iodine loading on the post accident SLCRS, Control Room Ventilation Intake, and ERF Ventilation Recirculation charcoal filters due to the long-term containment leakage resulting from atmospheric conversion, and associated compliance to the requirements of Regulatory Guide 1.52 (Reference 13).

5.3.2 Computer Codes

The QA Category 1 Stone & Webster computer codes utilized to support this application, are listed below. The referenced computer codes have been used extensively by S&W to support nuclear power plant design. With the exception of ARCON96, all of the computer codes listed below are a part of BVPS current licensing basis.

1. Industry Computer Code SCALE 4.3, "Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations And Personal Computers," Control Module SAS2H, Version 3.1, developed by ORNL (S&W Program NU-230, V05, L03).

2. S&W Proprietary Computer Program ACTIVITY2, "Fission Products in a Nuclear Reactor," NU-014, V01, L03.

3. S&W Proprietary Computer Program IONEXCHANGER, NU-009, V01, L02.

4. Industry Computer Code ARCON96, "Atmospheric Relative Concentrations in Building Wakes" developed by PNL (S&W Program EN-292, V00, L00).

5. S&W Proprietary Computer Code, PERC2, "Passive Evolutionary Regulatory Consequence Code," NU-226, V00, L01.

6. S&W Proprietary Computer Code, SWNAUA, "Aerosol Behavior in Condensing Atmosphere," NU-185, V02, L00.

7. S&W Computer Code, SW-QADCGGP, "A Combinatorial Geometry Version of QAD-5A," NU-222, V00, L02.

8. S&W Proprietary Computer Code, LOCTIC, "Loss of Coolant Transient Inside Containment," NU-100, V23, L03.

5.3.3 Radiation Source Terms: LOCA & CREA Analyses

5.3.3.1 Core Activity Inventory

The current rated licensed core power level for each of the BVPS Units is 2689 MWT. The fuel cycle length is 18 months.

In accordance with regulatory guidance, radiological evaluations for accident related issues must include an appropriate margin for power uncertainty. At BVPS, installation of improved feedwater measurement instrumentation used for the calorimetric power calculation allows the margin for instrument error to be reduced from the traditional 2% as recommended in Regulatory Guide 1.49 (Reference 14). The uncertainty allowance for calorimetric thermal power measurement has been reduced to 0.6%, as approved by the NRC in its SER for the BVPS-License Amendments No. 243/122 for Units 1/2, respectively (Reference 50). Consequently, the equilibrium core inventory utilized for accident analyses must reflect, at a minimum, a power level of 2705 MWT.

The inventory of fission products in the reactor core developed in support of atmospheric conversion is based on maximum full-power operation of the core at a power level of 2918 MWT, and current licensed values of fuel enrichment and burnup. This core inventory can therefore be utilized in accident analyses for uprated core power levels up to 2900 MWT. As discussed earlier, the higher power level is used to make this assessment applicable at future uprated conditions. Note that the assessment utilizes a common bounding core inventory to encompass both units.

The Beaver Valley cores are currently fueled with 157 Westinghouse 17x17 V5H fuel assemblies, which have Zirlo® cladding (note that Zircaloy assemblies are also enveloped by the analysis, since BVPS-1 will utilize several Zircaloy) and integral fuel burnable absorber (IFBA). Core designs are being used with Westinghouse "Robust Fuel Assembly" (RFA) designs, which also will use Zirlo® and IFBA. The BVPS-1 and BVPS-2 equilibrium core inventory calculations performed for the AST envelope both fuel designs. The core inventory has been determined using the isotope generation and depletion computer code ORIGEN S, and a methodology that will be not be significantly affected by future minor changes in the fuel management plan.

The licensing basis methodology utilized to develop the core inventory remains unchanged by this application.

The ORIGEN-S calculation is performed by utilizing the Control Module SAS2H of the ORNL SCALE 4.3 computer code package. SCALE 4.3 is a modular code system developed by Oak Ridge National Laboratory. SCALE 4.3 has been qualified by Stone & Webster for QA Category I use. The SAS2H control module provides a sequence to calculate the nuclide inventory in a fuel assembly by calling various neutron cross section treatment modules and the exponential matrix point-depletion module ORIGEN-S. SAS2H-ORIGENS calculates the time-

dependent neutron flux and the buildup of fissile transuranic nuclides. It accounts for all major nuclear interactions including fission, activation, and various neutron absorption reactions. It calculates the neutron-activated products, the actinides and the fission products in a reactor core.

The reactor core in each of the BVPS units consists of 157 fuel assemblies with various Uranium-235 enrichments. The current licensed maximum values for fuel enrichment are 5.0% in BVPS-1, and 4.85% in BVPS-2 (Ref. Technical Specification Tables 3.9-1). A LAR is being submitted to change BVPS-2 to 5% enrichment. To account for variation of U-235 enrichment in fresh fuel, the radionuclide inventories were calculated for a 4.2% average enriched core (representing minimum enrichment), and 5% average enriched core (representing maximum enrichment). The higher activity for each isotope from the above two enrichment cases is chosen to represent the inventory of that isotope in the equilibrium core.

The equilibrium core at the end of a fuel cycle is assumed to consist of fuel assemblies with three different burnups, i.e., approximately 1/3 of the core is subjected to one fuel cycle, 1/3 of the core to two fuel cycles and 1/3 of the core to three fuel cycles. Minor variations in fuel irradiation time and duration of refueling outages will have a slight impact on the estimated inventory of long-lived isotopes in the core. However, these inventory changes will have an insignificant impact on the radiological consequences of postulated accidents.

The maximum design fuel burnup for the fuel is up to 62,000 MWD/MTU (peak pin burnup limit), however the region average design burnup for the discharge region is 54,000 MWD/MTU (Reference 51). Because the BVPS units are currently operated on an 18-month refueling cycle, the period of irradiation has been nominally estimated at 518 days, which is of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.

The core inventory developed by ORIGEN-S using the above methodology includes over 800 isotopes. The BVPS equilibrium core inventory of dose significant isotopes relative to LWR accidents is presented in Table 5.3.3-1.

5.3.3.2 Coolant Activity Inventory

The licensing basis methodology utilized to develop the design basis and technical specification primary and secondary coolant inventory remain unchanged by this application.

Design Basis Primary and Secondary Coolant Activity Concentrations

Stone & Webster QA Category I proprietary computer code, ACTIVITY2, is used to calculate the design basis primary coolant fission product activity concentrations for both BVPS-1 and BVPS-2 based on the core inventory developed using ORIGEN-S and as discussed in Section 5.3.3.1. The source terms for the design basis primary coolant fission product activity include leakage from 1% defective fuel and decay of parent, grand parent and second parent. The

depletion terms of the primary coolant fission product activity include radioactive decay, purification of the letdown flow and neutron absorption when the coolant passes the reactor core. To encompass the different operating conditions, the coolant activity is calculated with and without Volume Control Tank purge. The nuclear library includes 3rd order decay chains of approximately 200 isotopes.

Stone & Webster QA Category I proprietary computer code, IONEXCHANGER, is used to calculate the design basis halogen and remainder activity concentrations in the secondary side liquid. The source terms for the secondary side activity include the primary-to-secondary leakage in steam generators and the decay products of parent, grand parent and second parent. The depletion terms of the secondary side liquid activity include radioactive decay, and purification due to the steam generator (SG) blowdown flow.

The design basis noble gas concentrations in the secondary steam are calculated by dividing the appearance rate (μCi/sec) by the steam flow rate (g/sec). The noble gas appearance rate in the SG steam space includes the primary-to-secondary leak contribution and the noble gas generation due to decay of halogens in the SG liquid. The activity concentrations of the other isotopes in the steam are determined by the SG liquid concentrations and the partition coefficients recommended in NUREG 0017, Rev 1 (Reference 15).

Technical Specification Primary and Secondary Coolant Activity Concentrations

The technical specifications Limiting Condition for Operation (LCO) 3.4.8a limit the specific activity for iodines in the reactor coolant to 0.35 μCi/gm Dose Equivalent I-131 (DEI) in Unit-2, and 0.1 μCi/gm DEI in Unit-1, respectively. Technical Specification LCO 3.4.8b for each unit limits the non-iodine nuclides that make up >95% of the gross primary coolant activity with half-lives greater than 15 minutes to 100 / $\bar{E}$ μCi/gm.

In accordance with BVPS current licensing basis, the estimated isotopic technical specification concentration of fission products in the reactor coolant is based on the design basis fission product mix, adjusted to reflect the more limiting of the failed fuel percentages associated with the two limits identified in Technical Specification LCO 3.4.8. This approach is reasonable as the mix of isotopes in the reactor coolant is determined by the leakage of core activity from the defective fuels and the escape coefficients of the isotopes and its precursors. The above is already factored into the design basis coolant isotopic mix, and the referenced mix is not expected to change (i.e., between iodine vs non-iodine isotopes) such that both Technical Specification LCOs co-exist at the same time.

The maximum allowable failed fuel percentage in the reactor coolant associated with Technical Specification LCO 3.4.8a, i.e., the technical specification for iodines in the reactor coolant, is developed using the thyroid dose conversion factors (DCF) for I-131, I-132, I-133, I-134, and I-135 as obtained from the BVPS-1 and BVPS-2 technical specification Section 1.19 and are

consistent with ICRP-30 (Reference 56). Average beta and gamma energies per disintegration are based on References 16, 17, 18 & 19.

The technical specification iodine dose equivalent I-131 concentrations per iodine nuclide in the reactor coolant are calculated with the following equation:

$$DEI-131(i)(\mu Ci/gm) = \frac{C(i) \times CT_{tot}}{\sum\{F(i) \times C(i)\}} \tag{5-1}$$

Where:

DCF(i)	=	ICRP 30 Thyroid Dose Conversion Factor per Nuclide (Rem/Ci)
F(i)	=	DCF(i)/DCF I-131
C(i)	=	reactor coolant equilibrium iodine concentration per nuclide (μCi/gm)
CT_{tot}	=	reactor coolant total (DE I-131) tech. spec. iodine concentrations (μCi/gm).

Note that per the BVPS technical specifications, CT_{tot} is 0.1 μCi/gm for BVPS-1 and 0.35 μCi/gm for BVPS-2. For the purposes of performing bounding analyses representative of both units, the coolant concentration based on the BVPS-2 technical specifications is utilized for the accident analyses.

The maximum allowable failed fuel percentage in the reactor coolant associated with Technical Specification 3.4.8b; i.e., the technical specification for non-iodine fission products in the reactor coolant, is calculated with the following equation:

$$\text{Non} - \text{Iodine Act}(j)\ \mu Ci/gm = \frac{100 \times C(j)}{\sum\{C(j) \times [E_\beta(j) + E_\gamma(j)]\}} \tag{5-2}$$

Where:

$E_\beta(j)$	=	average beta energy for isotope j (MeV/disintegration)
$E_\gamma(j)$	=	average gamma energy for isotope j (MeV/disintegration)
C(j)	=	reactor coolant equilibrium Non-Iodine Concentration per nuclide (μCi/gm)
100	=	Normalization factor [(μCi/gm) (MeV/disintegration)]

Based on the above, it is determined, that the failed fuel percentage associated with Technical Specification 3.4.8a, i.e., the specific activity limit for the iodines in the reactor coolant, is more limiting. In addition, it is determined that with respect to accident analyses, the mix associated with the no VCT purge case is more limiting.

The technical specification iodine dose equivalent I-131 concentrations per iodine nuclide in the secondary coolant are calculated using the same equation as provided earlier for the reactor coolant. The secondary liquid technical specification concentration CT_{tot} is 0.05 μCi/gm and 0.1 μCi/gm DE I-131 for BVPS-1 and BVPS-2, respectively, where C(i) is the design secondary coolant equilibrium concentrations per nuclide.

The noble gas and halogen primary and secondary coolant technical specification Activity Concentrations for BVPS-1 and BVPS-2 are presented in Table 5.3.3-2 and 5.3.3-3, respectively. As discussed earlier, for the purposes of performing bounding analyses representative for both units, the BVPS-2 values are utilized, as needed, for both the LOCA and the CREA.

5.3.4 Accident Atmospheric Dispersion Factors

5.3.4.1 Site Boundary Atmospheric Dispersion Factors

The exclusion area boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application. These values are presented in Table 5.3.4-1. As discussed earlier, for the purposes of performing bounding analyses representative for both units, the BVPS-2 EAB χ/Q's will be utilized in assessing the impact of atmospheric conversion on the LOCA and the CREA. As noted in Table 5.3.4-1, the LPZ χ/Q's are the same for both Units.

5.3.4.2 On-Site Atmospheric Dispersion Factors: LOCA & CREA Analyses

The control room χ/Q values for the environmental release paths associated with the LOCA and CREA, from each unit, are calculated using the latest version of the "Atmospheric Relative Concentrations in Building Wakes" (ARCON96) methodology (Ramsdell, 1997, Reference 20). The ERF χ/Q values for the environmental release paths associated with the LOCA, are also addressed. Stone & Webster has qualified computer code ARCON96 for QA Category I use. The input data consists of hourly on-site meteorological data, release characteristics (e.g., release height and stack flow rate), the cross-sectional building area affecting the release, and receptor information (e.g., distance and direction from the release to the control room air intake and intake height). The ARCON96 runs were developed in accordance with draft NRC guidance on control room habitability assessments (Reference 22).

ARCON96 methodology has the ability to evaluate ground-level, vent, and elevated stack releases and treats building wake effects and stable plume meander effects when applicable. This methodology is also able to evaluate diffuse and area source releases using the virtual point source technique, wherein initial values of the dispersion coefficients are assigned based on the size of the diffuse or area source. The various averaging period χ/Q values are calculated directly from running averages of the hourly χ/Q values.

The BVPS meteorological measurement program meets the requirements of Regulatory Guide 1.23 (Reference 54) and Regulatory Position C.1.1 of Regulatory Guide 1.145 (Reference 55), and is described in detail in the UFSARs, Chapter 2.2.3. A continuous representative 5-year period of hourly average data from the BVPS meteorological tower (i.e., January 1, 1990 through December 31, 1994) is used in this calculation. Each hour of data, at a minimum, had a validated wind speed and direction at the 10-meter level and a temperature difference between the 45- and 10-meter levels.

All releases are conservatively treated as ground-level as there are no releases at this site that are high enough to escape the aerodynamic effects of the plant buildings (i.e., 2.5 times Containment Building height, U.S. NRC, 1982). In addition, the stack/vent release flows are not necessarily maintained throughout the accident period. The applicable structure relative to building wake effects on the releases is based release/receptor orientation.

In support of the LOCA and the CREA dose analyses, the specific release-receptor combinations for BVPS-1 for which χ/Q values are calculated are as follows:

Release Points	On-Site Receptors
1. BVPS-1 Containment Edge	1. BVPS-1 Control Room Intake
2. BVPS-1 Containment Top (SLCRS)	2. BVPS-2 Control Room Intake
3. BVPS-1 Refueling Water Storage Tank (RWST) Vent	3. Control Room In-leakage
4. BVPS-1 Main Steam Relief Valves	4. BVPS-2 Auxiliary Building Northwest Corner
	5. BVPS-1 Service Building Roof Vent
	6. Emergency Response Facility (ERF) Normal Operation Intake
	7. ERF Edge Closest to Containment (Used for Post LOCA In-leakage)

The releases/receptor combinations for which χ/Q values are calculated are as follows:

- Release Points 1, 2, and 3 to Receptors 1 through 7
- Release Point 4 to Receptors 1 through 3

Similarly, the release-receptor combinations for BVPS-2 for which χ/Q values are calculated are as follows:

Release Points	On-Site Receptors
1. BVPS-2 Containment Edge	1. BVPS-1 Control Room Intake
2. BVPS-2 Containment Top (SLCRS)	2. BVPS-2 Control Room Intake
3. BVPS-2 Refueling Water Storage Tank (RWST) Vent	3. Control Room In-leakage
4. BVPS-2 Main Steam Relief Valves	4. BVPS-2 Auxiliary Building Northwest Corner
	5. BVPS-1 Service Building Roof Vent
	6. Emergency Response Facility (ERF) Normal Operation Intake
	7. ERF Edge Closest to Containment (Used for Post LOCA In-leakage)

The releases/receptor combinations for which χ/Q values are calculated are as follows:

- Release Points 1, 2, and 3 to Receptors 1 through 7
- Release Point 4 to Receptors 1 through 3

The following assumptions are made for these calculations:

- The plume centerline from each release is transported directly over each receptor (conservative).
- The applicable structure relative to quantifying building wake effects on the dispersion of the releases is based on release/receptor orientation relative to the plant structures.
- The distances from the BVPS-1 and BVPS-2 containment edges to the receptors are determined from the closest edge of the containment building (conservative).
- The release elevations are set equal to the receptor elevations in cases where the releases are not from a clearly defined point, such as the containment edge releases. Where both the release and receptor are not clearly defined points, both elevations are set equal to grade elevation (conservative).
- Only the containment edge release is considered to be a diffuse source as the release is from the entire containment surface. Diffuse source treatment allows the calculation of initial values of the dispersion coefficients (i.e., sigma y and sigma z). These values are determined by the height and width of the containment divided by a factor of six based on Reference 22 (i.e., sigma y = 6.83 m and sigma z = 7.44m). All other releases are conservatively treated as point sources.

- The control room air intake χ/Q values are representative of the worst case χ/Q values for control room unfiltered in-leakage purposes since the distances and directions from the release points to these receptors are very similar.

  Control room tracer gas tests have indicated that a potential source of unfiltered inleakage into the control room during the post accident pressurization mode are the normal operation dampers associated with the control room ventilation system to which it is reasonable to assign the same χ/Q as that of the Control Room air intake.

  The other source of potential inleakage is that associated with ingress/egress and leakage via door seals. This inleakage is assigned to the door leading into the control room that is considered the point of primary access. This door is located in-between the BVPS-1 and BVPS-2 control room air intakes and is located close enough to the air intakes to allow the assumption that the χ/Q associated with this source of inleakage would be reasonably similar to that associated with the air intakes.

- The χ/Q values at the ERF edge closest to the BVPS-2 containment is conservatively assumed to be representative of the post-accident χ/Q values to the Emergency Response Facility which includes the Technical Support Center (TSC) and the Emergency Operations Facility (EOF).

- The ARCON96 default wind direction range of 90°, centered on the direction that transports the gaseous effluents from the release points to the receptors is used in the calculation and is based on Draft Regulatory Guide DG-1111 (Reference 22).

- The ARCON96 values for surface roughness length (i.e., 0.20 meter) and sector averaging constant (4.3) are based Draft Regulatory Guide DG-1111 (Reference 22).

- All releases are treated as ground-level as there are no release conditions that merit categorization as an elevated release (i.e., 2.5 times containment building height, with respect to the BVPS configuration (conservative).

The BVPS-1 and BVPS-2 χ/Qs values for all release-receptor combinations utilized for the LOCA and CREA dose assessments are summarized in Tables 5.3.4-2 and 5.3.4-3, respectively.

5.3.5 Dose Calculation Methodology: LOCA & CREA Analyses

A S&W proprietary computer program, PERC2, is used to calculate the Committed Effective Dose Equivalent (CEDE) from inhalation and the Deep Dose Equivalent (DDE) from submersion due to halogens, noble gases and other nuclides at the offsite locations and in the control room. The CEDE is calculated using the Federal Guidance Report No.11, Sept. 1988 (Reference 23) dose conversion factors. The committed doses to other organs due to inhalation of halogens,

noble gas, other nuclides and their daughters are also calculated. PERC2 is a multiple compartment activity transport code with the dose model consistent with the regulatory guidance. The decay and daughter build-up during the activity transport among compartments and the various cleanup mechanisms are included.

The PERC2 activity transport model, first calculates the integrated activity (using a closed form integration solution) at the offsite locations and in the control room air region, and then calculates the cumulative doses as described below:

Committed Effective Dose Equivalent (CEDE) Inhalation Dose - The dose conversion factors by internal organ type are applied to the activity in the air space of the control room, or at the EAB/LPZ. The exposure is adjusted by the appropriate respiration rate and occupancy factors for the CR dose at each integration interval as follows:

$$Dh(j) = A(j) \times h(j) \times C2 \times C3 \times CB \times CO$$

Where:

Dh(j)	=	Committed Effective Dose Equivalent (rem) from isotope j
A(j)	=	Integrated Activity (Ci-s/m^3)
h(j)	=	Isotope j Committed Effective Dose Equivalent (CEDE) dose conversion factor (mrem/pCi) based on Federal Guidance Report No.11, Sept. 1988
C2	=	Unit conversion of $1x10^{12}$ pCi/Ci
C3	=	Unit conversion of $1x10^{-3}$ rem/mrem
CB	=	Breathing rate (m^3/s)
CO	=	Occupancy factor

Deep Dose Equivalent (DDE) from External Exposure - According to the guidance provided in Section 4.1.4 and Section 4.2.7 of RG 1.183, the Effective Dose Equivalent (EDE) may be used in lieu of DDE in determining the contribution of external dose to the TEDE if the whole body is irradiated uniformly. The EDE in the control room is based on a finite cloud model that addresses buildup and attenuation in air. The dose equation is based on the assumption that the dose point is at the center of a hemisphere of the same volume as the control room. The dose rate at that point is calculated as the sum of typical differential shell elements at a radius R. The equation utilizes, the integrated activity in the control room air space, the photon energy release rates per energy group from activity airborne in the control room, and the ANSI/ANS 6.1.1-1991 “neutron and gamma-ray flux-to-dose-rate factors” (Reference 24).

The Deep Dose Equivalent at the EAB and LPZ locations is very conservatively calculated using the semi-infinite cloud model outlined in TID-24190, Section 7-5.2, Equation 7.36, (Reference 25) where 1 rad is assumed to be 1 rem.

$$\gamma D\infty(x,y,0) \text{ rad} = 0.25\ E\gamma_{BAR}\ \psi(x,y,0) \qquad (5\text{-}3)$$

$E\gamma_{BAR}$	=	average gamma released per disintegration (Mev/dis)
$\psi(x,y,0)$	=	concentration time integral (Ci-sec/m^3)
0.25	=	$[\ 1.11 \times 1.6x10^{-6} \times 3.7x10^{10}\ ]\ /\ [\ 1293 \times 100 \times 2\ ]$

where:

1.11	=	ratio of electron densities per gm of tissue to per gm of air
$1.6x10^{-6}$ (erg/Mev)	=	number of ergs per Mev
$3.7x10^{10}$ (dis/sec-Ci)	=	disintegration rate per curie
1293 (g/m^3)	=	density of air at S.T.P.
100	=	ergs per gram per rad
2	=	factor for converting an infinite to a semi-infinite cloud

5.3.6 Radiological Accident Analyses

5.3.6.1 Introduction

As discussed in Section 5.3, the changes proposed by this application will potentially impact the reported dose consequences of the BVPS-1 and BVPS-2 LOCA, CREA, MSLB and the BVPS-1 LRA. The radiological consequences of the remaining design basis accidents are not adversely impacted by this application.

BVPS-1 and BVPS-2 LOCA and CREA:

As discussed in Section 5.1, the two accidents impacted by atmospheric conversion are the LOCA and the CREA. With this licensing application, the methodology/scenarios discussed in the BVPS 1 and BVPS-2 UFSARs for the LOCA and CREA site boundary and control room analyses are being updated to reflect the guidance provided in Regulatory Guide 1.183 (Reference 3).

Within the foreseeable future, BVPS 1 and 2 intend to submit a LAR to operate at an uprated level. To this end, the accident analyses supporting atmospheric conversion are conservatively

based on an equilibrium core and NSSS power level that is representative of uprated conditions; and reflect bounding parameter values to encompass an event at either unit.

In addition, the updated LOCA and CREA dose analyses reflect the updated inputs, as well as the changes included as part of the atmospheric conversion package and discussed in Section 5.1 of this application.

The worst 2-hour period dose at the EAB, and the dose at the LPZ for the duration of the release, is calculated for LOCA and the CREA based on postulated airborne radioactivity releases. This represents the post accident dose to the public due to inhalation and submersion for each of these events. In accordance with Reference 3, offsite breathing rates used are as follows: 0-8 hr (3.5E-04 m^3/sec), 8-24 hr (1.8E-04 m^3/sec), 1-30 days (2.3E-04 m^3/sec). Due to distance/plant shielding, the dose contribution at the EAB/LPZ due to direct shine from contained sources is considered negligible for all the accidents.

The 0 to 30-day dose to an operator in the control room due to airborne radioactivity releases is developed for both the LOCA and the CREA. This represents the post accident dose to the operator due to inhalation and submersion. The CR shielding design is based on the LOCA which represents the worst case DBA relative to radioactivity releases. The direct shine dose due to contained sources/external cloud is included in the CR doses reported for the LOCA.

BVPS-1 and BVPS-2 MSLB and BVPS-1 LRA

As discussed in Section 5.1, with the exception of the BVPS-1 and BVPS-2 Main Steam Line Break (MSLB) and the BVPS-1 Locked Rotor Accident (LRA), the remaining design basis events are not adversely impacted by the changes included with atmospheric conversion. The CREBAPS is currently credited for the BVPS-1 and BVPS-2 MSLB and the BVPS-1 LRA. In addition, the automatic CREBAPS initiation function of the control room monitors is credited in the BVPS-1 LRA

The dose impact of eliminating the above two design features on control room habitability following a BVPS-1 and BVPS-2 MSLB and a BVPS-1 LRA is assessed using existing licensing basis methodology / assumptions / regulatory acceptance criteria, and is not included in the selective application of AST methodology.

In accordance with current licensing basis, the control room is assumed to be pressurized at T=30 mins following a MSLB at either unit, based on manual initiation of the control room pressurization system. However, credit is taken for the CREVS and not CREBAPS to provide control room pressurization post accident.

The BVPS-1 LRA continues to depend on the control room radiation monitors to alarm on high radiation levels in the control room. However, instead of automatic initiation of CREBAPS, the updated analysis reflects control room pressurization at T=1 hr following a LRA at BVPS-1

based on manual initiation of CREVS upon acknowledgement of a high radiation signal from the referenced radiation monitors.

5.3.6.2 Control Room Design/Operation/Transport Model

BVPS is served by a single control room that supports both units. The joint control room is serviced by two ventilation intakes, one assigned to BVPS-1 and the other to BVPS-2. These air intakes are utilized for both the normal as well as the accident mode.

During normal plant operation, both ventilation intakes are operable and provide a total supply of 500 cfm of unfiltered outside air makeup which includes all potential inleakage and uncertainties.

Current plant design will automatically isolate the control room and initiate control room pressurization via the control room emergency bottled air pressurization system (CREBAPS) upon receipt of a containment high-high pressure signal (i.e., CIB) from either unit, or a high radiation alarm from the control room area monitors. The above signals will also initiate the time delay relay that delays start of the control room filtered emergency ventilation system (CREVS) for one hour which is intended to correspond to the depletion of the bottled air and the time for the containment to return to sub-atmospheric pressure.

One hour after actuation, the time delay relay will automatically start one of the trains of the BVPS-2 control room filtered emergency pressurization system fans. On detection of fan failure, the second train is automatically initiated after a short time delay.

In the unlikely event that neither of the BVPS-2 trains can be put in service, operator action may be utilized to initiate the BVPS-1 control room filtered emergency pressurization system. This unlikely scenario is utilized in accident analysis to allow flexibility in taking out a BVPS-2 CREVS train for maintenance.

With this application, BVPS will eliminate the accident mitigation function of CREBAPS and the sixty minute time delay associated with the CREVS. Also no credit is taken for the capability of the safety related control room radiation monitors to automatically initiate the control room emergency pressurization system.

The BVPS-1 and BVPS-2 CIB signals will initiate the BVPS-2 CREVS without a 60 minute time delay. As discussed above, in the event one of the BVPS-2 trains is out of service, and the second train fails to start, operator action will be utilized to initiate the BVPS-1 control room emergency pressurization system.

For purposes of dose assessment, no credit will be taken for automatic initiation of the control room emergency ventilation system following any design basis accident.

The CREVS is designed to maintain a positive pressure in the control room greater than 1/8 inch w.g. The CR emergency pressurization intake HEPA and charcoal filters are tested in accordance with the requirements of Regulatory Guide 1.52 (Reference 13) and Generic Letter 99-02 (Reference 52), respectively.

With this application, the BVPS-1 technical specifications will be revised to reflect the more limiting acceptance criteria noted for the HEPA and charcoal filters in the BVPS-2 technical specifications. In accordance with the margin requirements of the referenced regulatory guidance, the LOCA and the CREA analyses utilize an efficiency of 99% for particulates, and 98% for elemental and organic iodine. The estimated dose consequences of the remaining BVPS accidents are conservative in that they have not been updated to reflect the more stringent BVPS-2 acceptance criteria.

The control room ventilation recirculation flow is not filtered, but remains in service during emergency conditions to maintain the control room within design temperature limits.

The BVPS control rooms (CR) are contained in a single control room envelope, which is therefore modeled as a single region. Isotopic concentrations in areas outside the control room envelope are assumed to be comparable to the isotopic concentrations at the control room intake locations. To support development of bounding control room doses for the LOCA and the CREA, the most limiting χ/Q associated with the release point/receptor for an event in either unit is utilized.

The control room post-accident ventilation model utilized in the LOCA and CREA dose analysis corresponds to an assumed "single intake" which utilizes the worst case atmospheric dispersion factor (χ/Q) from release points associated with accidents at either unit, to the limiting control room intake. The atmospheric dispersion factors for the various combinations of release point/receptor applicable for a LOCA or a CREA, at BVPS-1 and BVPS-2, are provided in Tables 5.3.4-2 and 5.3.4-3, respectively.

In accordance with current licensing basis, the atmospheric dispersion factors associated with control room inleakage are assumed to be the same as those utilized for the control room intake. As discussed earlier in Section 5.3.4.2, control room tracer gas tests have indicated that potential sources of unfiltered inleakage into the control room during the post accident pressurization mode are the normal operation dampers associated with the control room ventilation system to which it is reasonable to assign the same χ/Q as that of the Control Room air intake.

The other source of potential inleakage is that associated with the door seals. This inleakage, plus an allowance of 10 cfm for ingress and egress, is assigned to the door leading into the control room that is considered the point of primary access. This door is located at grade level and in-between the BVPS-1 and BVPS-2 control room air intakes. It is located close enough to the referenced air intakes to allow the assumption that the χ/Q associated with this source of inleakage would be reasonably similar to that associated with the air intakes.

Taking into account Loss of Offsite Power (LOOP), the maximum estimated delay following a LOCA to achieve control room isolation after receipt of a CIB signal to switch from control room normal operation to emergency ventilation mode, is 77 seconds, which accounts for delays due to Emergency Diesel Generator start, sequencing and damper movement/re-alignment. CREVS Train A fan is expected to get a start signal at T=90 secs. Considering the CREVS time delay relay setting for the Train B fan start, plus fan acceleration time, the total auto start delay is estimated to be 137 sec.

Since the LOCA and the CREA analysis is intended to be bounding for an event at either unit, no credit is taken for automatic initiation of the BVPS control room emergency pressurization system. Rather it is assumed that operator action will be necessary to initiate the control room emergency filtered pressurization system, and that a pressurized control room will be available within T=30 mins.

The LOCA and CREA dose models conservatively assume that prior to achieving control room isolation, the unfiltered intake flow into the control room is equivalent to the intake associated with normal operation, i.e., 500 cfm.

A control room unfiltered inleakage of 300 cfm is conservatively assumed during the time it is isolated, (i.e., between T=77 secs to T=30 mins). This value is based on the results of tracer gas testing in the isolated mode, and includes a 10 cfm unfiltered inleakage due to ingress/egress as required by SRP 6.4 (Reference 26). The analysis takes into account measured inleakage using mean values of the tracer gas test measurements.

The control room emergency filtered ventilation intake flow varies between 600 to 1030 cfm, which includes allowance for measurement uncertainties. For the LOCA and the CREA, the control room unfiltered inleakage during the emergency pressurization mode is conservatively assumed to be 30 cfm (includes 10 cfm unfiltered inleakage due to ingress/egress) to reflect the results of tracer gas testing in the pressurized mode. The analysis takes into account measured inleakage using mean values of the tracer gas test measurements.

For reasons outlined below, the dose model uses the minimum intake flow rate of 600 cfm in the pressurized mode as it is considered to be more limiting. Although the intake of radioisotopes is higher at the larger intake rate of 1030 cfm, it is small compared to the radioactivity entering the control room, in both cases, due to unfiltered inleakage. Consequently, the depletion of airborne activity in the control room via the higher exhaust rate of 1030 cfm make the lower intake rate of 600 cfm more limiting from a dose consequence perspective. This argument holds true because the CEDE from inhalation is far more limiting than the DDE from immersion, which is principally from noble gases.

Table 5.3.6-1 lists key assumptions/parameters associated with BVPS control room design utilized for the LOCA and the CREA dose analyses.

5.3.6.3 Loss of Coolant Accident (LOCA)

RG 1.183 identifies the large break LOCA as the design basis case of the spectrum of break sizes for evaluating performance of release mitigation systems including containment, and facility siting relative to radiological consequences.

BVPS has identified four activity release paths following a LOCA: (a) Containment Vacuum System Release, (b) Containment Leakage (c) ESF System Leakage and d) RWST back leakage. Since BVPS design includes redundant safety related hydrogen recombiners, there is no requirement for post-LOCA containment purge for hydrogen control within 30 days of the LOCA.

Table 5.3.6-2 lists some of the key assumptions/parameters utilized to develop the radiological consequences following a LOCA at either unit. Parameter values are bounding and encompass the event occurring at either unit.

5.3.6.3.1 Doses due to Submersion and Inhalation

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a LOCA. PERC2 is a QA Category I code. It utilizes an exact solution analytical computational process that addresses radionuclide progeny, time dependent releases, transport rates between regions and deposition of radionuclide concentrations in sumps, walls and filters.

5.3.6.3.1.1 Containment Vacuum System Release

It is assumed that the containment Vacuum System is operating at the initiation of the LOCA and that the release is terminated as part of containment isolation. In accordance with Reference 3, the entire RCS inventory, assumed to be at technical specification levels, is released to the containment at T = 0 hours. It is conservatively assumed that 100% of the volatiles are instantaneously and homogeneously mixed in containment atmosphere. Containment pressurization (due to the RCS mass and energy release), combined with the relief line cross-sectional area, results in a 2200 scfm release of containment atmosphere (based on BVPS-1, equivalent release from BVPS-2 is 1600 scfm), to the environment over a period of 5 seconds (i.e., prior to containment isolation). Since the release is isolated within 5 seconds after the LOCA, i.e., before the onset of the gap phase release assumed to be at 30 seconds, no fuel damage releases are postulated.

Per Reference 3, the chemical form of the iodine released from the RCS is assumed to be 97% elemental and 3% organic. The containment Vacuum System line is routed to the Process Vent which is located on top of the BVPS-1 Cooling Tower. However, since the associated piping is non-seismic, it is conservatively assumed that the release occurs at containment wall.

No credit is taken for processing this release via the safety related ventilation exhaust and filtration system that services the areas contiguous to containment; i.e.; the Supplementary Leak Collection System (SLCRS) filters. To ensure bounding values, the atmospheric dispersion factors utilized for this release reflects the worst value between the containment wall release point and the SLCRS release point for 0-2 hr time period.

An assessment of the activity release via this pathway demonstrates that its contribution to the site boundary and control room dose is negligible.

5.3.6.3.1.2 Containment Leakage

The inventory of fission products in the reactor core available for release via containment leakage following a LOCA is based on Table 5.3.3-1 which represents a conservative equilibrium reactor core inventory of dose significant isotopes, assuming maximum full power operation at uprated conditions, and taking into consideration fuel enrichment and burnup.

The fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core. Containment sprays are utilized as one of the primary means of fission product cleanup following a LOCA. BVPS design includes a containment quench spray and a containment re-circulation spray system at each of the units. Following post LOCA containment pressurization, the quench spray system is automatically initiated by the CIB signal, and injects cooling water from the refueling water storage tank (RWST), into the containment, via the quench spray system spray headers. Based on an assumption of a LOOP coincident with the LOCA, the quench spray is assumed to be initiated, at either unit, by approximately T=85 secs, and is available until depletion of the RWST inventory based on maximum ESF. The recirculation spray system which is assumed to be initiated at T=720 secs (value based on BVPS-2, BVPS-1 value is 300 secs) takes suction from the containment sump and provides recirculation spray inside containment via the recirculation spray headers. Credit for recirculation spray is taken up to 4 days post-LOCA.

Mixing of the “effectively” sprayed volume of containment with the unsprayed volume of the containment also facilitates the cleanup. In order to quantify the effectiveness of the containment spray systems, the volume fraction of containment that is sprayed, and the mixing rate between the sprayed and unsprayed volumes are quantified.

Fission Product Cleanup

In the effectively sprayed region, fission product cleanup is actively accomplished by the quench and recirculation spray systems and passively by transport of particulates to the spray droplets and heat sink surfaces as a result of steam condensation on these surfaces. In the unsprayed region, only the passive gravitational settling phenomenon promotes particulate removal.

The analysis presented below envelopes both BVPS-1 and BVPS-2. All parameter values used in the analysis are conservative for both units.

Removal of Particulates by Sprays

The particulates are effectively removed from the containment atmosphere by the quench and recirculation spray systems. The particulate removal rate is calculated with Stone & Webster's proprietary SWNAUA Computer Program (Reference 27). The SWNAUA Program is a derivative of the NAUA/MOD4 Computer Program (Reference 28).

The NRC staff has previously reviewed the SWNAUA spray model in the recent Alternative Source Term application submitted by Fort Calhoun Station. In its Safety Evaluation Report (Reference 53) the staff stated, "In its review of the proposed spray removal rates, the staff reviewed the brief description of the model provided by OPPD, reviewed the analysis inputs and assumptions, evaluated the reasonableness of the estimated removal rates, and reviewed the use of the estimated removal rates in the radiological analyses." "…the staff finds that OPPD's modeling of containment spray removal is acceptable,…."

Although the SWNAUA computer code is a derivative of the NAUA/MOD4 computer code, the NAUA/MOD4 code does not include a model for aerosol removal by sprays. The aerosol removal model for sprays was developed and incorporated into the SWNAUA code by S&W as a conservative model suitable for Design Basis Accident calculations. The model correlation's that were implemented into SWNAUA tend to underestimate the spray removal coefficient. The spray model was originally described in Reference 29. For the effectively sprayed region of the containment, S&W employs only the conservatively developed spray removal model and conservative condensation rates for the diffusiophoresis calculation when performing DBA calculations. While agglomeration is considered in the calculation, its impact on the resulting particulate removal rates is negligible. In summary, the aerosol removal rates calculated by SWNAUA are conservative lower bound estimates.

There are several aerosol mechanics phenomena that promote the depletion of aerosols from the containment atmosphere. These include the natural phenomena of agglomeration, gravitational settling, diffusional plate-out, and diffusiophoresis; and removal by fluid mechanical interaction with the falling droplets that enter the containment atmosphere through the spray system nozzles. For Beaver Valley, the particulate removal calculation for the effectively sprayed region only takes credit for the removal effectiveness of diffusiophoresis and sprays. Agglomeration of the aerosol is considered. If gravitational settling and diffusional plate-out were considered, the spray removal coefficients would have been slightly reduced but the total removal effectiveness by all removal mechanisms would have increased. However, only gravitational settling of aerosols is credited in the unsprayed region.

The spray model in SWNAUA evaluates the particulate removal efficiency for each particle size in the aerosol by the following mechanisms: inertial impaction, interception, and Brownian

diffusion. The aerosol removal constant due to spray is presented in NUREG-0772 (Reference 30) as:

$$\lambda_{spray} = \frac{3\,F_m\,h\,\varepsilon}{4\,R_{sp}\,\rho_w\,V} \times \frac{v_{spray} - v_{sed}}{v_{spray}} \tag{5-4}$$

where

λ_{spray}	=	Particulate removal constant for spray,
F_m	=	Spray mass flow rate,
h	=	Spray fall height,
ε	=	Collision efficiency,
R_{sp}	=	Spray droplet radius,
ρ_w	=	Density of the spray droplet,
V	=	Effectively sprayed volume of containment,
v_{spray}	=	Velocity of the spray droplets, and
v_{sed}	=	Aerosol sedimentation velocity.

The collision efficiency is divided into three contributing mechanisms as described in BMI-2104 (Reference 31):

$$\varepsilon = \varepsilon_i + \varepsilon_r + \varepsilon_d \tag{5-5}$$

where

ε_i	=	Efficiency due to inertial impaction,
ε_r	=	Efficiency due interception and
ε_d	=	Efficiency due to Brownian diffusion.

For viscous flow around the spray droplet, the inertial impaction efficiency is given in NUREG-0772 (Reference 30):

$$\varepsilon_i = \frac{1}{\left[1 + \frac{0.75\,\ln\,(2\,Stk)}{Stk - 1.214}\right]^2} \tag{5-6}$$

The critical Stokes number, Stk, for viscous flow is 1.214; for Stk below this value, the model assumes the efficiency of inertial impaction is 0.0. The Stk is calculated from BMI-2104 (Reference 31):

$$Stk = \frac{2\,\rho_p\, r^2\, C_c\,(v_{spray} - v_{sed})}{9\,\mu\, R_{sp}} \tag{5-7}$$

where

r	=	Aerosol particle radius,
ρ_p	=	Aerosol density,
C_c	=	Cunningham slip correction factor,
μ	=	Gas viscosity.

For droplet sizes typical of nuclear plant spray systems, the data of Walton and Woolcock (Reference 32) show that collision efficiency will be closer to that predicted for potential flow around the droplet. Calvert (Reference 33) fitted this data to the expression:

$$\varepsilon_i = \left(\frac{Stk}{Stk + 0.7}\right)^2 \tag{5-8}$$

The collision efficiency predicted by this equation is always higher than that predicted by the viscous flow expression given above. The Calvert's fit is employed in this calculation.

As for the remaining constituents of the collision efficiency, the spray model employs an interception efficiency of the form:

$$\varepsilon_r \cong \frac{3}{2}\left(\frac{r}{R_{sp}}\right)^2 x \left(1 - \frac{1}{3}\frac{r}{R_{sp}}\right) \tag{5-9}$$

which is a conservative approximation of the expression given by BMI-2104 (Reference 31). The efficiency due to Brownian motion is also taken from this report:

$$\varepsilon_d = 3.5\, Pe^{-2/3} \tag{5-10}$$

where

Pe	=	Peclet number
	=	$2v_{spray}R_{sp}/D_B$,
D_B	=	Aerosol diffusion coefficient
	=	$k_{Boltz}TB$ (Fuchs, Reference 34, p. 181),
k_{Boltz}	=	Boltzmann constant
	=	1.3804×10^{-16} erg/K.
T	=	Temperature, K

Fuchs (Reference 34, p. 27) gives the aerosol mobility, B:

$$B = \frac{C_c}{6 \pi \mu r} \tag{5-11}$$

In most cases, the overall collision efficiency is dominated by inertial impaction, but for small aerosols, Brownian diffusion may become dominant. The collision efficiency due to inertial impaction increases as the aerosol size is increased, whereas that due to Brownian diffusion increases as the aerosol size decreases.

The model has the capability of handling a distribution of up to 20 droplet radii with the spray removal efficiency being determined for each aerosol size bin.

Elia and Lischer (Reference 29) investigated the use of a single spray droplet size in the analysis instead of a drop size distribution. While Reference 29 does not specifically analyze the Beaver Valley spray system, the parameter sensitivities for the spray model are applicable. The paper demonstrates that the droplet diameter distribution can be represented by a single diameter that is the mass mean diameter. The case 6 droplet distribution presented in the paper is for the Sprayco 1713A nozzle that is frequently used by the nuclear industry for fission product/heat removal spray systems. The paper notes that the mass mean droplet diameter for this nozzle is 1050μ. This diameter approximates that used in case 1, 1000μ. The spray flow rate used for both case 1 and case 6 is 10,000 gpm. Table 2 in the paper indicates that the spray removal rates for these two cases are very close. The mass mean spray droplet radius for Beaver Valley is specified in the table below.

The paper also investigated the variation of particulate removal coefficient with droplet diameter. Cases 1 through 3 vary the mass mean droplet diameter from 500μ to 1500μ. Although Table 2

in the paper indicates that these cases assume a spray flow of 10,000 gpm, the spray removal coefficient reduction by about 67 percent is expected to be independent of spray flow rate.

The bounding plant parameters for the Beaver Valley Units are listed below.

Plant Parameters for Fission Product Cleanup Calculations	
Parameter	**Value**
Sprayed Containment Volume	3.1973×10^{10} cm^3
Fall Height	2,403 cm
Spray Flow Rate	8,350 gpm (722-5,250 sec) 6,350 gpm (5,250-12,600 sec) 3,440 gpm (12,600-346,000 sec)
Spray droplet radius	350 : (722-18,000 sec) 518.5 : (18,000-346,000 sec)

The spray droplet radius/spray coverage fraction combination for BVPS-2 (518.5 :/80 percent) is bounded by the BVPS-1 combination (350 :/63 percent) up to 18,000 sec when the BVPS-2 quench spray is terminated..

The containment pressure, temperature, and humidity transients following the NUREG-1465 style DBA are presented in Table 5.3.6-3.

Description of Aerosol

The chemical composition of the aerosol is only important as it relates to the density of aerosol utilized in the development of spray lambdas. The chemical composition during the gap release phase is assumed to be pre-dominantly CsOH. The chemical composition during the in-vessel release phase is assumed to be 20 percent CsOH, 20 percent indium, and 60 percent silver. These assumed compositions are based on a review of the SASCHA experimental results. The aerosol input data for SWNAUA are provided below.

Description of Aerosol	
Minimum Aerosol Radius	1.0000E-07 cm
Maximum Aerosol Radius	1.0000E-02 cm
Maximum Number of Aerosol Size Bins	100
From 30 sec to 1830.0 sec	
Aerosol Injection Rate	6.13000E+00 (gm/sec)
Mean Geometric Radius	7.50000E-06 cm
Geometric Standard Deviation	1.560
Aerosol Density	3.700 gm/cc
From 1830 sec to 6510.0 sec	
Aerosol Injection Rate	5.85300E+01 (gm/sec)
Mean Geometric Radius	4.00000E-05 cm
Geometric Standard Deviation	1.460
Aerosol Density	4.600 gm/cc

Removal of Particulates by Diffusiophoresis

During diffusiophoresis, particulate matter is entrained in the steam as it flows to the condensation surfaces. In this calculation, steam is assumed to condense on the spray droplets, on particulate matter, and on heat sinks. The diffusiophoresis model in the SWNAUA computer code is the same as that in the NAUA/MOD4 computer code.

The steam condensation rates used by SWNAUA are calculated by the LOCTIC computer code. The LOCTIC code has been used to calculate the conservative DBA containment pressure and temperature responses on several applications and has been accepted by NRC. LOCTIC pressure and temperature results have been shown to be in close agreement with the NRC's CONTEMPT-LT computer code by confirmatory analyses performed by the staff.

Because LOCTIC predicts a conservatively high containment pressure transient, the rate of steam condensation from the containment atmosphere is minimized. The steam condensation rates that are input to the diffusiophoresis calculation of SWNAUA are calculated as LOCTIC's prediction of the steam removal rate from the containment atmosphere. The condensation rates are given in Table 5.3.6-3.

The coefficient for removal of particulates from the effectively sprayed and unsprayed regions of the containment are plotted versus time in Figures 5.3.6-1 and 5.3.6-2, respectively. For the effectively sprayed region, the aerosol removal is due to sprays and diffusiophoresis. The particulate removal coefficient in the unsprayed region is due to gravitational settling only.

Removal of Elemental Iodine

Studies done as part of Stone & Webster's Source Term Project concluded that at least some of the elemental iodine would deposit on the aerosol particles due to the high quantity of aerosol released.

During spray operation, the elemental removal rates for the sprays always exceeds 20 hr^{-1}, the maximum value permitted by NUREG-0800, Standard Review Plan Section 6.5.2 (Reference 35). For Beaver Valley, the calculated particulate removal coefficients for the sprays varies from values below this NRC limit to values that are much greater. Thus, during spray operation the elemental iodine is conservatively assumed to be removed by sprays at the same rate as the aerosol particles when the aerosol removal rate is lower than 20 hr^{-1} and at 20 hr^{-1} when the aerosol removal rate is calculated to be higher than the NRC limit.

A plateout removal coefficient for elemental iodine has been calculated with the model provided in NUREG-0800, Standard Review Plan Section 6.5.2 (Reference 35). In the effectively sprayed region, a minimum plateout coefficient of 2 hr^{-1} is calculated for Beaver Valley.

No credit is taken for elemental iodine removal in the unsprayed region.

Effectively Sprayed Containment Volume Fraction

The sprayed volume fraction of the containment is determined by superimposing the spray patterns onto the containment arrangement drawings. The sprayed volume is the volume of the unblocked spray patterns. The spray patterns are based on the nozzle manufacturer's laboratory tests at atmospheric conditions. The patterns have been compressed to account for the higher density atmosphere that exists during the DBA. The effectively sprayed volume is calculated by combining the highly mixed unsprayed regions with the sprayed.

The effective spray coverage fraction is assessed to be 63.0 percent of the containment free volume. The concentration of fission products is expected to be uniform in the containment volume above the operating floor since this volume is open with very few obstructions to mixing. The sprayed volume is taken as the free volume above the operating floor plus the volume below the operating floor that is covered by the recirculation spray. Actually, the whole containment is expected to be uniform in fission product concentration based on the following discussion, but the sprayed volume fraction has been limited to 63.0 percent.

Containment Mixing

Mixing in the containment following a postulated Design Basis Accident (DBA) results from four mechanisms: 1) momentum transfer from the fluid jet exiting the break; 2) momentum transfer from the spray droplets to the surrounding gas; 3) forced and natural convection flows within the containment atmosphere; and 4) molecular diffusion. All of these mechanisms will work together to enhance mixing within the containment to provide a homogeneous gas mixture and prevent local accumulation of fission products.

Good containment compartment mixing will occur during the blowdown period of the postulated DBA due to the break effluent. The momentum of the jet from the break will cause turbulent mixing within the containment. This was demonstrated in a test performed for a high velocity jet source (Reference 36). Results from this test showed that "when the jet was initiated, local gas velocities, even far from the source, increased by a factor of three to five times over background velocities caused by natural convection and fan-induced recirculation." Although this test was performed for an ice condenser lower compartment geometry, the test results would be applicable to subcompartments (e.g., steam generator cubicle, pressurizer cubicle) which are open to the containment.

Forced convection in the containment atmosphere will be generated by the containment spray systems that are designed to cool the containment atmosphere. The spray will induce mixing by imparting momentum to the containment atmosphere. Air entrainment by the spray causes bulk mass motion that creates both large- and small-scale turbulence. Therefore, complete mixing should occur within a few minutes following a DBA with containment spray operation (Reference 37).

In addition, steam condensation and cooling of the containment atmosphere by the sprays will result in flow to low pressure regions. This does not result in significant mixing within individual compartments, although significant intercompartment fluid transfer can occur (Reference 38).

Natural convection due to density differences (buoyant effects) is another source of mixing in the containment atmosphere. Gas flow occurs whenever there is a temperature difference between the wall and the bulk atmosphere. Gases heated or cooled by the walls will rise or fall, respectively, due to the density differences between the gas and the surrounding atmosphere. This buoyant force imparts momentum to the gas and significant turbulent mixing will result.

The presence of large heat sinks in the containment, such as internal walls, together with localized heat sources, such as hot equipment surfaces, will be expected to set up large-scale natural circulation cells. These circulation cells will help decrease any stratification that may occur in areas with the absence of jet induced or forced convection flows. Tests conducted during the containment systems experiment (CSE) program in a steam/air atmosphere indicated that natural convection caused good mixing in a large vessel (References 39 and 40).

After completion of the blowdown period of the postulated DBA, natural convection flows within the containment atmosphere will also be developed due to additional break effluent. The ECCS injects cooling water into the reactor cold leg. The injected water will exit the break as steam/water mixture. Buoyancy forces will cause the released steam to rise. This upward steam flow will generate containment mixing due to the entrainment of the atmosphere gases in the steam plume. The extent of mixing in areas away from the break due to the buoyant thermal plume discharging into the containment is a function of geometry, plume to atmosphere density ratio, and ratio of momentum to buoyancy forces (Reference 38).

Molecular diffusion is another mechanism that would provide mixing within the containment following a postulated DBA. Diffusion occurs due to concentration gradients. The rate of diffusion is too slow to result in mixing of large containment volumes in short times by itself, although molecular diffusion would add to the other mixing processes previously discussed.

The containment internal structures are designed to be as open as practical to allow the circulation and mixing mechanisms to function. The volume above the operating floor, which comprises the majority of the containment volume, does not have significant barriers to obstruct mixing from the various mechanisms.

The steam generator and pressurizer subcompartments, the annulus between the crane wall and containment wall, and the hoisting spaces are open at the top (near the top for the upper pressurizer cubicle) and bottom and connect with each other at various elevations. Extensive use is made of gratings at intermediate levels within the compartments. This design arrangement enhances mixing by establishing air movements and flow paths.

The internal design of the containment structures allows air to circulate freely. All cubicles and compartments within the containment are provided with openings near the top as well as openings in the floor to allow air circulation. Convective mixing in conjunction with containment spray ensures a uniform fission product concentration in the containment.

Containment system experiment tests (References 40 & 41) have verified that adequate mixing of the containment atmosphere is achieved by the containment spray system.

For Beaver Valley, the mixing rate between the effectively sprayed volume and the unsprayed volume of the containment is assumed to be 2 hr^{-1}, the rate permitted by NUREG-0800, the Standard Review Plan Section 6.5.2.

Radiological Transport Model

As indicated previously, the fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core. In accordance with Regulatory Guide 1.183, two fuel release phases are considered for DBA analyses: (a) the gap release, which begins 30 seconds after the

LOCA and continues for 30 mins and (b) the early In-Vessel release phase which begins 30 minutes into the accident and continues for 1.3 hours.

Per Reference 3, the core inventory release fractions, by radionuclide groups, for the gap and early in-vessel damage are as follows:

Group	Gap Release Phase	Early In-Vessel Release Phase
Noble gas	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium Group	-	0.05
Ba, Sr	-	0.02
Noble Metals	-	0.0025
Cerium Group	-	0.0005
Lanthanides	-	0.0002

The elements in each Radionuclide Group released to the containment following a LOCA are assumed to be as follows (note that the groupings were expanded from that in RG 1.183 to address isotopes in the core with similar characteristics.

Noble gases: Xe, Kr, Rn, H

Halogens: I, Br

Alkali Metals: Cs Rb

Tellurium Grp: Te, Sb, Se, Sn, In, Ge, Ga, Cd, As, Ag

Ba,Sr: Ba, Sr, Ra

Noble Metals: Ru, Rh, Pd, Mo, Tc, Co

Cerium Grp: Ce, Pu, Np, Th, U, Pa, Cf, Ac

Lanthanides: La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am, Gd, Ho, Tb, Dy

Since the BVPS long term sump pH is controlled to values of 7 and greater, the chemical form of the radioiodine released from the fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine. With the exception of noble gases, elemental and organic iodine, all fission products released are assumed to be in particulate form.

The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase. The release into the containment is assumed to terminate at the end of the early in-vessel phase, approximately 1.8 hours after the LOCA.

In the "effectively" sprayed region the activity transport model takes credit for aerosol removal due to steam condensation and via containment recirculation and quench sprays based on spray flowrates associated with minimum ESF. It considers mixing between the sprayed and unsprayed regions of the containment, reduction in airborne radioactivity in the containment by concentration dependent aerosol removal lambdas, and isotopic in-growth due to decay.

Since, using SRP 6.5.2 methodology, the calculated elemental iodine spray removal lambdas are greater than 20 hr^{-1}, it is conservatively assumed that the sprays remove the elemental iodine at the same rate as the aerosols when the aerosol removal rates are less than 20 hr^{-1}, and at 20 hr^{-1} when the aerosol removal rate is greater than 20 hr^{-1}. In the effectively sprayed region, a minimum plateout coefficient of 2 hr^{-1} is calculated for BVPS. This allows a maximum elemental iodine removal rate in the effectively sprayed region, during the spray period, of 22 hr^{-1}.

In the unsprayed region, the aerosol removal lambdas reflect gravitational settling. No credit is taken for elemental iodine removal in the unsprayed region.

Since the spray removal coefficients are based on calculated time dependent airborne aerosol mass, there is no restriction on the DF for particulate iodine. The maximum DF for elemental iodine is based on SRP 6.5.2 and is limited to a DF of 200.

Mixing between the "effectively" sprayed and unsprayed regions of the containment is assumed for the duration of the accident. As discussed earlier, though higher mixing rates are expected, the dose analysis conservatively assumes a mixing rate of 2 unsprayed volumes per hour in accordance with the default value noted in SRP 6.5.2.

Current BVPS design includes chemical addition into the quench spray system which ensures a long term sump pH equal or greater than or equal to 7.0. Long-term production of acids (HC1 and HNO_3), by irradiation is included in determining the long term sump pH. Long-term retention of iodine in sump liquids is strongly dependent on the sump pH. The dose analysis does not address iodine re-evolution as a sump pH of ≥ 7 is achieved well within 16 hours after the LOCA and is maintained for the duration of the accident. The definition of long term as it relates to sump pH and iodine re-evolution post LOCA is based on NUREG/CR 5732 (Reference 42).

Radioactivity is assumed to leak from both the sprayed and unsprayed region to the environment at the containment technical specification leak rate for the first day, and half that leakage rate for the remaining duration of the accident (i.e., 29 days). No credit is taken for processing the containment leakage via the safety related ventilation exhaust and filtration system that services

the areas contiguous to containment; i.e.; the Supplementary Leak Collection System (SLCRS) filters. To ensure bounding values, the atmospheric dispersion factors utilized for the containment release path reflects the worst value between the containment wall release point and the SLCRS release point for each time period.

5.3.6.3.1.3 ESF and RWST Back-Leakage

With the exception of noble gases, all the fission products released from the core in the gap and early in-vessel release phases are assumed to be instantaneously and homogeneously mixed in the primary containment sump water at the time of release from the fuel. The minimum sump volume increases to a steady state minimum value of 480,750 gallons two hours after the LOCA. Three sump volume values are utilized in the transport model. Up to the first half hour after the LOCA, the sump volume is about 14% of the final value. For the next one and a half hours the sump volume is about 43% of the final value. For the remainder of the accident the steady state minimum sump volume is utilized. In accordance with Regulatory Guide 1.183, with the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase. The subsequent environmental radioactivity release is discussed below:

- ESF leakage: Equipment carrying sump fluids and located outside containment are postulated to leak at twice the expected value of 5700 cc/hr (BVPS-1 value, BVPS-2 value is 2134 cc/hr) into the auxiliary building. ESF leakage is expected starting at initiation of the recirculation spray which at BVPS-1 is being revised to 300 seconds (start time based on maximum ESF) and at BVPS-2 is 720 seconds. Note that due to the long term nature of this release, minor variations in the start time of this release will not significantly impact the resultant doses. The dose analysis assumes that recirculation is initiated at 300 secs. The peak sump water temperature after 300 seconds is 250ºF. As noted in Regulatory Guide 1.183, the fraction of total iodine in the liquid that becomes airborne should be assumed to be equal to the fraction of the leakage that flashes to vapor. The flash fraction, (using Regulatory Guide 1.183 methodology,) associated with this temperature is calculated to be less than 10%. Consequently, in accordance with Regulatory Guide 1.183, 10% of the halogens associated with this leakage is assumed to become airborne and are exhausted (without mixing and without holdup) to the environment via the SLCRS vent located on top of Containment. In accordance with Regulatory Guide 1.183, the chemical form of the iodine released from the sump water is 97% elemental and 3% organic. No credit is taken for the SLCRS filters.

- RWST Back-leakage: Sump water back-leakage into the RWST (located in the Yard) is postulated to occur at twice the expected leakrate of 1 gpm, to be released directly to the environment via the RWST vent. Sump water begins to leak into the RWST at 36.43 minutes after the LOCA. At 86.3 minutes, the iodine begins to flow out of the RWST and disperses to the environment. A significant portion of the iodine associated with the RWST back-leakage is retained within the tank due to equilibrium iodine

distribution balance between the RWST gas and liquid phases (i.e., a time dependent iodine partition coefficient). Environmental airborne iodine activity resulting from RWST leakage is assumed to be 97% elemental and 3% organic. In the dose model, this phenomenon is modeled using a series of effective environmental release rate lambdas from the RWST vent. These release rates are provided in Figure 5.3.6-3. The analysis of RWST back-leakage envelops the RWST design modifications for ECCS switchover level setpoint and maximum temperature.

5.3.6.3.1.4 Accident Specific Control Room Model Assumptions – Inhalation and Submersion

Due to the rapid pressure transient expected following a LOCA, the Containment Isolation Phase B (CIB) signal which initiates the CR isolation and emergency ventilation following a LOCA is assumed to actuate at T=0 hours.

The analysis assumes a LOOP at T=0 hours. The impact of a LOOP at a more unfavorable time following the accident, such as during the fuel release phase, is not addressed per NRC Information Notice 93-17 (Reference 43). The need to evaluate a design basis event assuming a simultaneous or subsequent LOOP is based on the cause/effect relationship between the two events (an example illustrated in IN 93-17 is that a LOCA results in a turbine trip and a loss of power generation to the grid, thus causing grid instability and a LOOP a few seconds later, i.e., a reactor trip could result in a LOOP). IN 93-17 concludes that plant design should reflect all credible sequences of the LOCA/LOOP, but states that a sequence of a LOCA and an unrelated LOOP (which would be the case if a LOOP was assumed to occur 1 to 2 hours after the event) is of very low probability and is not a concern.

The remaining CR parameters utilized in this model is discussed in Section 5.3.6.2.

5.3.6.3.1.5 Site Boundary Dose Assessment

To find the “worst case 2-hour release window” for the EAB dose, the integrated dose versus time for each of the four pathways (i.e., containment pressure relief release path, containment leakage, ESF leakage, RWST back-leakage) is developed at increments of 0.1 hr. The 0-2 hr EAB Atmospheric Dispersion Factor is utilized for all cases. The analysis demonstrates that the maximum dose occurs with a two-hour release period that ends at approximately 2.5 hrs.

The EAB and LPZ dose following a LOCA is presented in Section 5.3.6.7.

5.3.6.3.2 Control Room Dose due to Direct Shine from the External Cloud and Contained Sources:

The dose contribution in the control room due to direct shine from the external cloud and from contained sources (for both bulk shielding and through penetrations), is addressed. The external

cloud contribution includes containment leakage, ESF leakage and RWST back-leakage. The contained sources include shine from the containment structure (skyshine dose is insignificant due to associated soft gammas, and the 2 ft concrete roof/walls of the control room), cable spreading room airborne source below the BVPS-2 portion of the combined control room through floor penetrations, cable tray mezzanine airborne source below the BVPS-1 portion of the combined control room through floor penetrations, control room emergency ventilation intake filters, and direct shine from the radiation source inside the RWST.

The maximum control room operator dose following a LOCA at either unit, based on atmospheric conversion, AST methodology and uprated conditions, is presented in Section 5.3.6.7.

5.3.6.4 Control Rod Ejection Accident (CREA)

S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a CREA.

This assessment follows the guidance provided in RG 1.183 for the CREA. Table 5.3.6-4 lists some of the key assumptions/parameters utilized to develop the radiological consequences following the CREA.

In accordance with guidance provided in RG 1.183, two independent release paths to the environment are analyzed: first, via containment leakage of the fission products released due to the event from the primary system to containment, assuming that the containment pathway is the only one available; and second, via releases from the secondary system, outside containment, following primary-to-secondary leakage in the steam generators, assuming that the latter pathway is the only one available.

The actual doses resulting from a postulated CREA would be a composite of doses resulting from portions of the release going out via the containment building and, portions via the secondary system. If regulatory compliance to dose limits can be demonstrated for each of the scenarios, the dose consequence of a scenario that is a combination of the two will be encompassed by the more restrictive of the two analyzed scenarios.

The BVPS CREA analysis evaluates the following two scenarios.

Scenario 1: The failed/melted fuel resulting from a postulated CREA is released into the RCS, which is released in its entirety into the containment via the ruptured control rod drive mechanism housing, is mixed in the free volume of the containment, and then released at containment technical specification leak rate. Environmental releases are assumed to occur via the containment wall.

Scenario 2: The failed/melted fuel resulting from a postulated CREA is released into the RCS which is then transmitted to the secondary side via steam generator tube leakage. The condenser is assumed to be unavailable due to a loss of offsite power. Environmental releases occur from the steam generators via the MSSVs and the ADVs.

In accordance with Regulatory Guide 1.183, a LOOP is assumed at the worst time, i.e., at T=0 hrs.

For the CREA event, the extent of fuel damage was calculated by Westinghouse, using previously approved accident analysis methodology, as currently described in the UFSAR. The fission product inventory of each of the damaged fuel rods was determined by dividing the total core inventory by 41,448, the number of fuel rods in the core. To account for differences in power level across the core, a design radial peaking factor of 1.75 was applied in determining the inventory of the damaged rods. The CREA results in less than 10% of failed fuel and 0.25% melted fuel.

In accordance with Regulatory Guide 1.183, the gap activity is assumed to be composed of 10% of the core noble gas and 10% of the core halogens associated with the percentage of fuel that has failed. The release fractions for the CREA are taken from Note 11 to Table 3 of Regulatory Guide 1.183. The gap release fractions listed in Table 3 were determined to be acceptable for use with currently approved LWR fuel having a peak burnup up to 62,000 MWD/MTU. This burnup is valid, provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burn-ups exceeding 54,000 MWD/MTU.

Depending on the release pathway, the composition of the melted fuel is varied. For the containment leakage pathway, the melted fuel activity released is assumed to be composed of 100% of the core noble gas and 25% of the core halogens associated with the percentage of fuel that has melted. For the Secondary System Release pathway, the melted fuel activity released is composed of 100% of the core noble gas and 50% of the core halogens associated with the percentage of fuel that has melted.

The chemical composition of the iodine in the gap/melted fuel is assumed to be 95% CsI, 4.85% elemental and 0.15% organic. However, because the sump pH is not controlled following a CREA, it is conservatively assumed that the iodine released via the containment leakage pathway has the same composition as the iodine released via the secondary system release pathway; i.e., it is assumed that for both scenarios, 97% of all halogens available for release to the environment are elemental, while the remaining 3% is organic.

Scenario 1: Transport from the Containment

The failed/melted fuel activity released due to a CREA into the RCS is assumed to be instantaneously released into the containment where it mixes homogeneously in the containment free volume. The containment is assumed to leak at the technical specification leak rate of

0.001 day^{-1} for the first 24 hours and at half that value for the remaining 29 days after the event. Except for decay, no credit is taken for depleting the halogen (or noble gas) concentrations airborne in the containment. No credit is taken for processing the containment leakage via the safety related ventilation exhaust and filtration system that services the areas contiguous to containment; i.e.; the Supplementary Leak Collection System (SLCRS) filters. To ensure bounding values, the atmospheric dispersion factors utilized for the containment release path reflects the worst value between the containment wall release point and the SLCRS release point for each time period.

Scenario 2: Transport from the Secondary System

The failed or melted fuel activity released due to a CREA into the RCS is assumed to be instantaneously and homogeneously mixed in the reactor coolant system and transmitted to the secondary side via primary to secondary steam generator (SG) tube leakage, assumed to be at the technical specification value of 150 gpd (@STP) from each steam generator (450 gpd total). The primary to secondary leakage terminates at 2500 seconds after the event when the primary pressure is below the secondary pressure. At BVPS, the SG tubes remain covered for the duration of the event; therefore, per Regulatory Guide 1.183, the gap/fuel iodines have a partition coefficient of 100 in the SG. The gap noble gases are released freely to the environment without retention in the SG.

The condenser is assumed unavailable due to a coincident loss of offsite power. Consequently, the radioactivity release resulting from a CREA is discharged to the environment from the steam generators via the MSSVs and the ADVs. The SG releases continue until shutdown cooling is initiated via operation of the RHR system (8 hrs after the accident) and environmental releases are terminated.

Per the regulatory requirement, the 2-hour EAB dose must reflect the “worst case” 2-hour activity release period following the CREA. The worst 2-hr EAB dose will occur during the initial 2 hour period, because the primary to secondary leakage stops at 0.6944 hours and the steam release rate is also the highest during this period.

The activity associated with the release of secondary steam/liquid, and primary to secondary leakage of normal operation RCS, (both at the technical specification limits) via the MSSVs/ADVs are insignificant compared to the failed fuel release, and are therefore not included in this assessment.

Accident Specific Control Room Model Assumptions

The analysis assumes that the CIB signal is not actuated, and the control room emergency ventilation system is manually initiated at T=30 min. after the CREA. The remaining Control Room parameters utilized in this model is discussed in Section 5.3.6.2.

The EAB, LPZ and Control room dose following a CREA are presented in Section 5.3.6.7.

5.3.6.5 Main Steam Line Break

As discussed in Section 5.1, one of the changes included with atmospheric conversion is the elimination of the CREBAPS for accident mitigation functions. CREBAPS is currently credited for the BVPS-1 and BVPS-2 MSLB.

The dose impact of eliminating the above design feature on control room habitability following a BVPS-1 and BVPS-2 MSLB is assessed using existing licensing basis methodology / assumptions, and regulatory acceptance criteria as noted in NUREG 0800, SRP 6.4 (Reference 26).

BVPS-1 MSLB

The existing design basis analysis supporting the control room doses following a BVPS-1 MSLB is based on manual initiation of CREBAPS at 30 minutes after the accident. The CREBAPS supplies the required pressurization air for 60 minutes. At T= 90 minutes, the control room ventilation is switched to CREVS which maintains control room pressurization using filtered intake air. The analysis assumes that operation of CREVS results in the pressurization flow being filtered with an effective removal efficiency of 97% for the iodine species.

As part of this application, operator action will be assumed to initiate CREVS instead of CREBAPS; i.e. the updated analysis assumes CREVS operation from T=30 mins to the end of the event.

Review of the current control room doses based on CREBAPS operation with subsequent operation of CREVS, indicate that the thyroid dose is the limiting dose (80% of the allowed dose of 30 rem for the co-incident spike case, 35% of the allowed dose for the pre-incident spike case). The external EDE and the skin DE are approximately three orders of magnitude less than the allowable regulatory limits.

The control room thyroid dose depends on the amount of intake of iodine into the control room and the depletion rate of the iodine activity existing in the room. Since the activity release period is 0 – 8 hours, and the χ/Q value remains unchanged during this time period, the iodine intake of the control room is a function of the activity release rate, intake flow rate, filter efficiency and the unfiltered inleakage rate.

During the co-incident spiking period (0 – 4 hr), the activity release from the faulted SG increases as a function of time after the accident. This activity increase is considered in evaluating the iodine intake into the control room for operations with CREBAPS, and without CREBAPS. The only depletion mechanism of the iodine activity in the control room is the exhausting of the air and decay.

A conservative scaling factor is developed based on the intake activity with CREBAPS operation and the intake activity without CREBAPS operation. This scaling factor is applied to the calculated thyroid dose to obtain an approximate upper bound value for the case that the CREBAPS / CREVS operation is replaced by the CREVS operation.

The control room dose following a MSLB at BVPS-1 is presented in Section 5.3.6.7.

BVPS-2 MSLB

The existing design basis analysis supporting the control room doses following a BVPS-2 MSLB is based on manual initialization of CREBAPS at 30 minutes after the accident. The CREBAPS supplies the required pressurization air for 60 minutes. At T= 90 minutes, the control room ventilation is switched to CREVS which maintains control room pressurization using filtered intake air. The analysis conservatively assumes that operation of CREVS results in the pressurization flow being filtered with an effective removal efficiency of 95% for the iodine species.

As part of this application, operator action will be assumed to initiate CREVS instead of CREBAPS; i.e.; the updated analysis assumes CREVS operation from T=30 mins to the end of the event.

Review of the current control room doses based on CREBAPS operation indicate that the thyroid dose is the limiting dose (9.3% of the allowed dose of 30 rem for the co-incident spike case, 4.1% of the allowed dose for the pre-incident spike case). The gamma dose and the beta dose are more than three orders of magnitude below the allowable regulatory limits.

As discussed earlier for BVPS-1, the control room thyroid dose depends on the amount of intake of iodine into the control room and the depletion rate of the iodine activity existing in the room. Since the activity release period is 0 – 8 hours and the χ/Q value remains unchanged during this time period, the iodine intake of the control room is a function of the activity release rate, intake flow rate, filter efficiency and the unfiltered inleakage rate.

During the co-incident spiking period (0 – 4 hr), the activity release from the faulted SG increases as a function of time after the accident. This activity increase is considered in evaluating the iodine intake into the control room for operations with CREBAPS, and without CREBAPS. The only depletion mechanism of the iodine activity in the control room is the exhausting of the air and decay.

A conservative scaling factor is developed based on the intake activity with CREBAPS operation, and the intake activity without CREBAPS operation. This scaling factor is applied to the calculated thyroid dose to obtain an approximate upper bound value for the case that the CREBAPS / CREVS operation is replaced by the CREVS operation.

The control room dose following a MSLB at BVPS-2 is presented in Section 5.3.6.7.

5.3.6.6 Locked Rotor Event (BVPS-1)

As discussed in Section 5.1, the changes included with atmospheric conversion include elimination of the CREBAPS for accident mitigation functions, and credit for elimination of the automatic initiation capability of the control room radiation monitors to initiate the control room pressurization system; specifically the CREBAPS. Both of these above features are currently credited for the BVPS 1 LRA.

The dose impact of eliminating the above design features on control room habitability following a BVPS-1 LRA is assessed using existing licensing basis methodology / assumptions, and regulatory acceptance criteria as noted in NUREG 0800, SRP 6.4 (Reference 26).

The updated analyses reflect continued dependence on the control room radiation monitors to alarm on high radiation levels in the control room. However, instead of automatic initiation of CREBAPS, the updated analysis reflects control room pressurization at T=1 hr following a LRA at BVPS-1 based on manual initiation of CREVS upon acknowledgment of a high radiation signal from the referenced radiation monitors.

The control room dose following a LRA at BVPS-1 is presented in Section 5.3.6.7.

5.3.6.7 Summary of Results

In support of atmospheric conversion, the dose consequences at the site boundary and the control room following design basis accidents that result in pressure transients in the containment, (i.e., the LOCA and the CREA), have been re-analyzed using alternative source terms, and taking into consideration post accident containment leakage for the duration of the accident.

In accordance with Regulatory Guide 1.183, the “worst 2-hour period” dose at the EAB, and the dose at the LPZ “for the duration of the release” is presented in Table 5.3.6-5. These dose values represent the post accident dose to the public due to inhalation and submersion for each of these two events. Due to distance and plant shielding, the dose contribution at the EAB/LPZ due to direct shine from contained sources is negligible for all accidents. The associated regulatory limits discussed in Section 5.3.1 are also presented.

Per regulatory guidance the control room dose is integrated over 30 days. The calculated doses address the fact that for events with a duration less than thirty days, the control room dose needs to include the remnant radioactivity within the control room envelope after the event is terminated, unless credit can be taken for a control room purge after the releases are terminated. The 30-day integrated dose to the control room operator, due to inhalation and submersion, is presented in Table 5.3.6-6 for both the LOCA and the CREA.

The control room shielding design is based on the LOCA, which represents the worst case DBA relative to radioactivity releases. The dose contribution due to direct shine from post LOCA contained sources as well as the external cloud are identified and included in the control room doses reported for the LOCA.

Note that this assessment concludes that neither the SLCRS charcoal nor HEPA filters are required to maintain the site boundary and control room doses following a LOCA or a CREA within the regulatory requirements.

It is concluded that the site boundary and control room dose consequences of the LOCA and the CREA remain within the regulatory requirements of 10CFR50.67 following atmospheric conversion.

In addition, as a result of the impact of the changes included with the containment conversion modification, the MSLB (both units) and the LRA (BVPS-1) have been re-evaluated. The dose impact of eliminating the previously discussed design features (see Sections 5.3.6.5 and 5.3.6.6) on control room habitability is assessed for the MSLB (both units) and LRA (BVPS-1) using existing licensing basis methodology / assumptions, and regulatory acceptance criteria as noted in NUREG 0800, SRP 6.4.

The following is concluded that if the T=30 – 90 minutes CREBAPS operation is replaced by the CREVS operation following a MSLB outside containment:

- For BVPS-1, the control room co-incident spike thyroid CDE will increase from 21.4 Rem to approximately 22.1 Rem, and the pre-incident spike thyroid CDE will increase from 10.4 Rem to approximately 11.0 Rem. All other control room doses will remain well below the regulatory limits. The increase in thyroid dose is less than 10% of the regulatory acceptance criteria and is therefore considered insignificant.

- For BVPS-2, the control room co-incident spike thyroid dose will increase from 2.78 Rem to approximately 2.88 Rem, and the pre-incident spike thyroid dose will increase from 1.22 Rem to approximately 1.32 Rem. All other control room doses will remain well below the regulatory limits. The increase in thyroid dose is less than 10% of the regulatory acceptance criteria and is therefore considered insignificant.

The updated BVPS-1 LRA analyses reflecting control room pressurization at T=1 hr following a LRA at BVPS-1 based on manual initiation of CREVS upon acknowledgment of a high radiation signal from the referenced radiation monitors results in doses of 3.23 Rem thyroid, 0.0174 Rem whole body, and 0.268 Rem skin. As expected, there is no change in the EAB or LPZ results from the previous analysis. The increase in thyroid dose from 3.1 Rem to 3.23 Rem thyroid is less than 10% of the acceptance criterion of 30 Rem and is therefore considered insignificant.

It is concluded that the control room doses for the MSLB (both units) and the LRA (BVPS-1) will continue to remain within the regulatory limits of SRP 6.4 following this application.

5.3.7 Other Radiological Assessments

5.3.7.1 Introduction

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 Technical Information Document 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 including establishing habitability of the Emergency Response Facility

- 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).

It is noted that the changes included with atmospheric conversion have no impact on radiological equipment qualification and vital access doses.

5.3.7.2 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 uprated 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 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. The NRC staff is assessing the effect of increased cesium releases (associated with AST methodology) on EQ doses as a generic safety

issue (GSI). Pending resolution of this issue, licensees may use either the AST or TID 14844 assumptions for performing the required EQ analyses.

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, and to assess the effect of varying the SLCRS ventilation exhaust rate per cubicle on airborne integrated gamma doses (refer to Section 6.3.1.2.2). Containment leakage was prorated between the contiguous areas based on the number, type, and size of the penetrations, as well as the administrative limits for specific Type B penetrations. Minimum cubicle ventilation exhaust rates were determined based on the acceptance criteria that

- for contiguous area cubicles with harsh environments, the airborne gamma dose due to atmospheric conversion would not increase the previously evaluated integrated dose by more than 5%, and

- for contiguous area cubicles with mild environments, the airborne gamma dose due to atmospheric conversion would not cause the cubicle to change its classification from a mild to a harsh environment.

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. For the most part, 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 Technical Information Document (normal plus accident) reported in the BVPS-1 EQ Radiation Zone Tables for the following areas:

- Elevation 768'7" of the Auxiliary Building: Zones J (General Area) and I (SLCRS cubicle)
- Elevation 752'6" of the Auxiliary Building: Zone H (General Area)
- Elevation 713' 6"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 Technical Information Document (normal plus accident) reported in the BVPS-2 EQ Radiation Zone Tables for the following areas:

- Elevation 773'6" of the Auxiliary Building: Zone 80 (SLCRS cubicle)
- Elevation 735'6" 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 BVPS-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 7.1.5.

5.3.7.3 Vital Access including ERF Habitability

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 44) 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 53), 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.

5.3.7.3.1 Vital Area Access

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. Airborne levels in the plant due to long term containment leakage resulting from atmospheric conversion were 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 BVPS-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 as discussed in Section 5.3.6, 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 (Reference 45) 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' 6."

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 22Rem / 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 773'6"of the Auxiliary Building, whereas Route 14 involves passage on the floor below, i.e., El 755' 6."

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.

5.3.7.3.2 ERF Habitability

Except as noted below, habitability of the BVPS ERF (which houses both the Technical Support Center (TSC) and the Emergency Operations Facility (EOF)) is assessed using methodology/accident release parameters similar to that utilized for the control room (i.e., Regulatory Guide 1.183). The assessment demonstrates compliance with paragraph IV.E.8 of Appendix E, to 10 CFR Part 50, Emergency Planning and Preparedness for Production and Utilization Facilities, regulatory guidance provided in Supplement 1 of NUREG-0737 (Reference 12) and the current BVPS licensing basis, without taking credit for operation of the ventilation equipment in the ERF.

<u>Doses due to Submersion and Inhalation</u>

Beaver Valley Power Station is served by a single Emergency Response Facility (ERF) that supports both units. During normal plant operation, the ERF ventilation intake flow of 3800 cfm (+/-10% for uncertainty) is processed through a HEPA filter with 99% efficiency. Unfiltered inleakage during normal operation is estimated at 2090 cfm.

Following a LOCA, the ERF is manually isolated and switched to an emergency filtered recirculation mode at T=30 mins. The ERF emergency ventilation recirculation flowrate is maintained at its design flow rate of 3800 cfm (+/-10%). The ERF emergency mode charcoal filters have a 90% removal efficiency relative to elemental and organic iodine, whereas the HEPA filters are rated at a 98% removal efficiency for particulates. The unfiltered inleakage into the ERF during the emergency mode is estimated to be 910 cfm, which includes 10 cfm for ingress and egress.

However, for the purposes of demonstrating habitability, no credit is taken for the ERF structure/normal or emergency ventilation systems. Because the facility is located a sufficient distance away from the BVPS-1 and 2 containment buildings, the atmospheric dispersion characteristics of potential activity releases following a LOCA are very favorable, and therefore, no ventilation design features are required to ensure habitability. The habitability analysis of the

ERF following a LOCA is performed by assuming that there is no ERF structure (i.e., the ERF is modeled as a point in the environment). Breathing rates and occupancy factors utilized are similar to those used for the control room. Table 5.3.7-1 lists key parameters associated with the ERF design.

ERF Dose due to Direct Shine from External Sources:

The dose contribution in the ERF due to direct shine from the external sources include shine from the Containment Structure (including skyshine), the ERF normal operation ventilation intake filter and the ERF emergency ventilation recirculation filters. Though not credited in the airborne dose assessment, the direct shine from the filters is included to address the potential of their usage post accident. The external cloud shine dose is included in the submersion dose described earlier. The dose from radiation sources inside the RWST is negligible because the elevation of the top of the source is below the elevation of state highway Route 168.

30 day Dose in the ERF.

The maximum 30 day dose to the operator in the ERF following a LOCA at either unit, based on atmospheric conversion, AST methodology and uprated conditions, is calculated to be 3 Rem TEDE, which includes 2.5 Rem TEDE due to submersion and inhalation. Note that this assessment concludes that neither the SLCRS filters, nor the ERF filters are required to maintain ERF Habitability.

5.3.7.4 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 discussed in Section 5.3.6, 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 ($5x10^{-3}$per 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
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.”

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	$5.77X10^{-4}$	$1.78X10^{-5}$

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 BVPS-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. The evaluations also demonstrate compliance with the post-accident environmental qualification requirements for class 1E electrical equipment in accordance with 10CFR50.49.

Normal Operation

An atmospheric containment during normal operation will not impact BVPS-1 and BVPS-2 compliance with the operator exposure limits of 10CFR20, or with the public exposure limits of 10CFR50 Appendix I.

Post Accident Site Boundary and Control Room Doses

To address the radiological impact of eliminating the current post-accident design basis requirement of achieving sub-atmospheric conditions inside the containment within 1 hour following accidents that result in containment pressurization, BVPS-1 and BVPS-2 are voluntarily revising/replacing the accident source term used in the 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).

The design basis accidents that result in pressure transients in containment are the Loss of Coolant Accident (LOCA) and the Control Rod Ejection Accident (CREA). The referenced analyses have been evaluated using the guidance provided in Regulatory Guide 1.183. The

estimated dose consequences at the EAB, LPZ and Control Room remain within the acceptance criteria of 10CFR50.67, as supplemented by Regulatory Guide 1.183 and SRP 15.0.1.

It is noted that the dose consequences from a LOCA and a CREA were conservatively based on the uprated core and NSSS power level and reflect bounding parameter values to encompass an event at either unit.

In addition, the updated LOCA and CREA dose analyses reflect the following updated inputs, as well as the changes included as part of the atmospheric conversion package.

Changes

- The analyses reflect the change in the RWST low level transfer setpoints which impacts the time the Emergency Core Cooling System (ECCS) switchover from the injection to the recirculation mode since this change could potentially impact the timing of the start of iodine releases due to post-LOCA RWST back leakage.

- The analyses reflect the change in the RWST temperature range since this change impacts the iodine release fraction via the RWST vent due to post-LOCA RWST back leakage.

- The analyses reflect the elimination of the control room bottled air emergency pressurization system (CREBAPS). This design feature is no longer considered necessary, since the BVPS containments will no longer be required to return to sub-atmospheric conditions within sixty minutes following a LOCA.

- The analyses address the elimination of the 60 minute time delay upon receipt of the CIB signal which is currently used to delay the start of the control room emergency ventilation system (CREVS) pressurization fans to accommodate initial control room pressurization via the bottled air system for one hour.

- The analyses will credit the revised Technical Specification acceptance criteria for the BVPS-1 control room HEPA and charcoal filters, which will make the BVPS-1 acceptance criteria similar to the more limiting criteria currently listed for the BVPS-2 control room filters.

- The analyses reflect not crediting the HEPA and charcoal filters in the Supplementary Leak Collection and Release System (SLCRS) exhaust flow for accident mitigation functions.

- The automatic initiation feature of the control room monitors to initiate the control room emergency pressurization system is not credited in the analysis. Note that this design feature is currently not credited in the LOCA and CREA analyses. Therefore this change has no impact on the dose consequences of a LOCA or a CREA.

Updated Inputs

- The analyses utilize conservative estimates of control room unfiltered inleakage that envelope the results of tracer gas testing performed in the year 2001.

- The analyses utilize updated control room atmospheric dispersion factors using the ARCON96 methodology.

- The analyses conservatively reflect manual initiation of CREVS. The control room is assumed to be pressurized via manual initiation of the CREVS within T=30 mins after the LOCA or CREA.

- The LOCA analysis does not take credit for the HEPA filter in the Emergency Response Facility (ERF) normal operation ventilation intake flow, or the HEPA and charcoal filters in the ERF post accident ventilation recirculation flow.

It is noted that the design basis of the remaining accidents (with the exception of the FHA, which is already based on 10CFR50.67) will continue to be 10CFR100.11 as documented in the current UFSARs.

The dose consequences of the remaining design basis events are not impacted by atmospheric conversion. However, some of the changes included with the containment conversion will impact the dose consequences of the following accidents. The CREBAPS is currently credited for the BVPS-1 and BVPS-2 MSLB and the BVPS-1 LRA. In addition, the automatic CREBAPS initiation function of the control room radiation monitors is credited in the BVPS-1 LRA.

The dose impact of eliminating the above two design features on control room habitability following a BVPS-1 and BVPS-2 MSLB and a BVPS-1 LRA is assessed using existing licensing basis methodology / assumptions. The estimated dose consequences remain within the regulatory acceptance criteria noted in NUREG 0800, SRP 6.4.

Equipment Qualification

The assessment of the impact of long-term post LOCA containment leakage resulting from atmospheric conversion and the associated use of AST methodology on the post-LOCA component of the Technical Information Document supporting equipment qualification in areas impacted by the design change concluded 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

- the increased activity accumulation on the SLCRS and control room filters at BVPS-1 and BVPS-2 indicate that radiation levels due to the accident contribution will result in an increase in the Technical Information Document (normal plus accident) reported in the BVPS EQ Radiation Zone Tables for the following areas:

  BVPS-1

  - Elevation 768'7" of the Auxiliary Building: Zones J (General Area) and I (SLCRS cubicle)
  - Elevation 752'6" of the Auxiliary Building: Zone H (General Area)
  - Elevation 713' 6"of the Control Building: Air Conditioning Equipment Room (BVPS-1 Control Room Filter Cubicle)

  BVPS-2

  - Elevation 773'6" of the Auxiliary Building: Zone 80 (SLCRS cubicle)
  - Elevation 735'6" 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 BVPS-2 EQ zones have been reviewed to ensure that they are qualified to the updated levels. The impact of atmospheric conversion on equipment qualification is addressed in Section 7.1.5.

Post-LOCA Vital Assess and ERF Habitability

It is concluded that there is no impact of long-term post LOCA containment leakage resulting from atmospheric conversion and the associated use of the AST methodology on the post-LOCA operator exposure while performing vital functions in areas impacted by the design change.

The estimated dose consequences at the ERF remain within the acceptance criteria of 10CFR50.67, as supplemented by Regulatory Guide 1.183. Note that this assessment concludes that neither the SLCRS filters nor the ERF filters are required to maintain ERF habitability.

Iodine Loading on Post-Accident Charcoal Filters

The assessment of the impact on iodine loading associated with the post-accident ventilation charcoal filters (i.e., the Supplementary Leak Collection and Release System (SLCRS) filter, Control Room Post Accident Ventilation Air Intake filter, Emergency Response Facility Post Accident Ventilation Recirculation filter) demonstrates continued compliance with the

acceptance criteria of Regulatory Guide 1.52 (Reference 13) relative to the allowable iodine loading on post accident filters.

5.5 References

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Table 5.3.3-1
BVPS-1 and BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWT)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
AG-111			5.05E+06		PU-239			2.86E+04
	PARENT:	AG-111M	5.06E+06			PARENT:	NP-239	1.66E+09
	GRAND PARENT:	PD-111	5.04E+06			GRAND PARENT:	U-239	1.66E+09
AG-112			2.28E+06		PU-240			3.87E+04
	PARENT:	PD-112	2.27E+06			PARENT:	NP-240	4.32E+06
AM-241			1.17E+04		PU-241			1.13E+07
	PARENT:	PU-241	1.13E+07		PU-242			2.01E+02
BA-137M			9.35E+06			PARENT:	AM-242	7.04E+06
	PARENT:	CS-137	9.81E+06		RB-86			1.69E+05
	GRAND PARENT:	XE-137	1.46E+08		RB-88			5.57E+07
BA-139			1.41E+08			PARENT:	KR-88	5.43E+07
	PARENT:	CS-139	1.37E+08			GRAND PARENT:	BR-88	2.99E+07
	GRAND PARENT:	XE-139	1.01E+08		RB-89			7.26E+07
BA-140			1.42E+08			PARENT:	KR-89	6.75E+07
	PARENT:	CS-140	1.23E+08			GRAND PARENT:	BR-89	2.08E+07
	GRAND PARENT:	XE-140	7.06E+07		RB-90			6.69E+07
BA-142			1.21E+08			PARENT:	KR-90	7.24E+07
	PARENT:	CS-142	5.48E+07			GRAND PARENT:	BR-90	1.13E+07
	GRAND PARENT:	XE-142	1.07E+07			2ND PARENT:	RB-90M	2.11E+07
BR-82			3.02E+05		RB-90M			2.11E+07
	PARENT:	BR-82M	2.62E+05			PARENT:	KR-90	7.24E+07
BR-83			9.37E+06			GRAND PARENT:	BR-90	1.13E+07
	PARENT:	SE-83M	4.69E+06		RH-103M			1.26E+08
	2ND PARENT:	SE-83	4.42E+06			PARENT:	RU-103	1.26E+08
BR-85			1.95E+07		RH-105			8.16E+07

Table 5.3.3-1 (Continued)
BVPS-1 and BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWT)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
CE-141			1.30E+08			PARENT:	RH-105M	2.53E+07
	PARENT:	LA-141	1.29E+08			GRAND PARENT:	RU-105	8.90E+07
	GRAND PARENT:	BA-141	1.28E+08			2ND PARENT:	RU-105	8.90E+07
CE-143			1.21E+08		RH-105M			2.53E+07
	PARENT:	LA-143	1.20E+08			PARENT:	RU-105	8.90E+07
CE-144			9.82E+07			GRAND PARENT:	TC-105	8.76E+07
CM-242			4.22E+06		RH-106			5.13E+07
	PARENT:	AM-242	7.04E+06			PARENT:	RU-106	4.63E+07
CM-244			5.97E+05		RU-103			1.26E+08
	PARENT:	AM-244	1.89E+07			GRAND PARENT:	MO-103	1.24E+08
CS-134			1.57E+07		RU-106			4.63E+07
	PARENT:	CS-134M	3.69E+06			2ND PARENT:	SN-125M	1.20E+06
CS-134M			3.69E+06		SB-127			6.92E+06
CS-135M			4.39E+06			PARENT:	SN-127	2.78E+06
CS-136			4.97E+06			2ND PARENT:	SN-127M	3.76E+06
CS-137			9.81E+06		SB-129			2.52E+07
	PARENT:	XE-137	1.46E+08			PARENT:	SN-129	9.90E+06
	GRAND PARENT:	I-137	7.47E+07			2ND PARENT:	SN-129M	9.29E+06
CS-138			1.48E+08		SB-130			8.37E+06
	PARENT:	XE-138	1.36E+08		SB-130M			3.47E+07
	GRAND PARENT:	I-138	3.80E+07			PARENT:	SN-130	2.61E+07
CS-139			1.37E+08		SB-131			6.09E+07
	PARENT:	XE-139	1.01E+08			PARENT:	SN-131	2.24E+07
	GRAND PARENT:	I-139	1.83E+07		SB-132			3.67E+07
CS-140			1.23E+08			PARENT:	SN-132	1.81E+07
	PARENT:	XE-140	7.06E+07		SB-133			5.08E+07
	GRAND PARENT:	I-140	4.81E+06		SE-83			4.42E+06

Table 5.3.3-1 (Continued)
BVPS-1 and BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWT)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	PARENT:	SM-155	3.11E+06		SM-153			4.02E+07
EU-156			2.29E+07			PARENT:	PM-153	7.37E+06
	PARENT:	SM-156	1.93E+06		SN-127			2.78E+06
EU-157			2.41E+06		SR-89			7.61E+07
H-3			4.36E+04			PARENT:	RB-89	7.26E+07
I-129			2.86E+00			GRAND PARENT:	KR-89	6.75E+07
	PARENT:	TE-129	2.40E+07		SR-90			7.21E+06
	GRAND PARENT:	TE-129M	4.87E+06			PARENT:	RB-90	6.69E+07
	2ND PARENT:	TE-129M	4.87E+06			GRAND PARENT:	KR-90	7.24E+07
I-130			2.07E+06			2ND PARENT:	RB-90M	2.11E+07
	PARENT:	I-130M	1.10E+06		SR-91			9.50E+07
I-131			7.78E+07			PARENT:	RB-91	8.85E+07
	PARENT:	TE-131	6.54E+07			GRAND PARENT:	KR-91	4.98E+07
	GRAND PARENT:	TE-131M	1.57E+07		SR-92			1.01E+08
	2ND PARENT:	TE-131M	1.57E+07			PARENT:	RB-92	7.83E+07
I-132			1.14E+08			GRAND PARENT:	KR-92	2.66E+07
	PARENT:	TE-132	1.12E+08		SR-93			1.14E+08
	GRAND PARENT:	SB-132	3.67E+07			GRAND PARENT:	KR-93	9.04E+06
I-133			1.60E+08		SR-94			1.14E+08
	PARENT:	TE-133	8.66E+07			GRAND PARENT:	KR-94	4.18E+06
	GRAND PARENT:	SB-133	5.08E+07		TC-99M			1.29E+08
	2ND PARENT:	TE-133M	7.12E+07			PARENT:	MO-99	1.45E+08
I-134			1.77E+08			GRAND PARENT:	NB-99	8.50E+07
	PARENT:	TE-134	1.41E+08		TC-101			1.33E+08
	2ND PARENT:	I-134M	1.59E+07			PARENT:	MO-101	1.33E+08
I-135			1.52E+08		TC-104			1.05E+08
I-136			6.99E+07			PARENT:	MO-104	9.99E+07

Table 5.3.3-1 (Continued)
BVPS-1 and BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWT)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
KR-83M			9.46E+06		TC-105			8.76E+07
	PARENT:	BR-83	9.37E+06			PARENT:	MO-105	7.38E+07
	GRAND PARENT:	SE-83M	4.69E+06		TE-127			6.81E+06
KR-85			8.27E+05			PARENT:	TE-127M	1.13E+06
	PARENT:	KR-85M	1.95E+07			GRAND PARENT:	SB-127	6.92E+06
	GRAND PARENT:	BR-85	1.95E+07			2ND PARENT:	SB-127	6.92E+06
	2ND PARENT:	BR-85	1.95E+07		TE-127M			1.13E+06
KR-85M			1.95E+07			PARENT:	SB-127	6.92E+06
	PARENT:	BR-85	1.95E+07			GRAND PARENT:	SN-127	2.78E+06
KR-87			3.91E+07		TE-129			2.40E+07
	PARENT:	BR-87	3.09E+07			PARENT:	TE-129M	4.87E+06
KR-88			5.43E+07			GRAND PARENT:	SB-129	2.52E+07
	PARENT:	BR-88	2.99E+07			2ND PARENT:	SB-129	2.52E+07
KR-89			6.75E+07		TE-129M			4.87E+06
	PARENT:	BR-89	2.08E+07			PARENT:	SB-129	2.52E+07
KR-90			7.24E+07			GRAND PARENT:	SN-129	9.90E+06
	PARENT:	BR-90	1.13E+07		TE-131			6.54E+07
LA-140			1.46E+08			PARENT:	SB-131	6.09E+07
	PARENT:	BA-140	1.42E+08			GRAND PARENT:	SN-131	2.24E+07
	GRAND PARENT:	CS-140	1.23E+08			2ND PARENT:	TE-131M	1.57E+07
LA-141			1.29E+08		TE-131M			1.57E+07
	PARENT:	BA-141	1.28E+08			PARENT:	SB-131	6.09E+07
LA-142			1.26E+08			GRAND PARENT:	SN-131	2.24E+07
	PARENT:	BA-142	1.21E+08		TE-132			1.12E+08
	GRAND PARENT:	CS-142	5.48E+07			PARENT:	SB-132	3.67E+07
LA-143			1.20E+08			GRAND PARENT:	SN-132	1.81E+07
MO-99			1.45E+08		TE-133			8.66E+07

Table 5.3.3-1 (Continued)
BVPS-1 and BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWT)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	PARENT:	NB-99M	5.82E+07			PARENT:	TE-133M	7.12E+07
	2ND PARENT:	NB-99	8.50E+07			GRAND PARENT:	SB-133	5.08E+07
MO-101			1.33E+08			2ND PARENT:	SB-133	5.08E+07
NB-95			1.34E+08		TE-133M			7.12E+07
	PARENT:	ZR-95	1.33E+08			PARENT:	SB-133	5.08E+07
	GRAND PARENT:	Y-95	1.28E+08		TE-134			1.41E+08
	2ND PARENT:	NB-95M	1.52E+06		XE-131M			1.08E+06
NB-95M			1.52E+06			PARENT:	I-131	7.78E+07
	PARENT:	ZR-95	1.33E+08			GRAND PARENT:	TE-131M	1.57E+07
	GRAND PARENT:	Y-95	1.28E+08		XE-133			1.60E+08
NB-97			1.27E+08			PARENT:	I-133	1.60E+08
	PARENT:	NB-97M	1.19E+08			GRAND PARENT:	TE-133M	7.12E+07
	GRAND PARENT:	ZR-97	1.26E+08			2ND PARENT:	XE-133M	5.05E+06
	2ND PARENT:	ZR-97	1.26E+08		XE-133M			5.05E+06
NB-97M			1.19E+08			PARENT:	I-133	1.60E+08
	PARENT:	ZR-97	1.26E+08			GRAND PARENT:	TE-133M	7.12E+07
ND-147			5.22E+07		XE-135			4.84E+07
	PARENT:	PR-147	5.18E+07			PARENT:	I-135	1.52E+08
	GRAND PARENT:	CE-147	4.92E+07			2ND PARENT:	XE-135M	3.36E+07
NP-239			1.66E+09		XE-135M			3.36E+07
	GRAND PARENT:	PU-243	4.23E+07			PARENT:	I-135	1.52E+08
	2ND PARENT:	U-239	1.66E+09		XE-137			1.46E+08
PD-109			3.26E+07			PARENT:	I-137	7.47E+07
PM-147			1.38E+07		XE-138			1.36E+08
	PARENT:	ND-147	5.22E+07			PARENT:	I-138	3.80E+07
	GRAND PARENT:	PR-147	5.18E+07		Y-90			7.49E+06
PM-148			1.41E+07			PARENT:	SR-90	7.21E+06

Table 5.3.3-1 (Continued)
BVPS-1 and BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWT)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	PARENT:	PM-148M	2.37E+06			GRAND PARENT:	RB-90	6.69E+07
PM-148M			2.37E+06		Y-91			9.87E+07
PM-149			4.82E+07			PARENT:	SR-91	9.50E+07
	PARENT:	ND-149	3.02E+07			GRAND PARENT:	RB-91	8.85E+07
	GRAND PARENT:	PR-149	2.80E+07			2ND PARENT:	Y-91M	5.51E+07
PM-151			1.60E+07		Y-91M			5.51E+07
	PARENT:	ND-151	1.58E+07			PARENT:	SR-91	9.50E+07
PR-142			5.57E+06			GRAND PARENT:	RB-91	8.85E+07
PR-143			1.18E+08		Y-92			1.02E+08
	PARENT:	CE-143	1.21E+08			PARENT:	SR-92	1.01E+08
	GRAND PARENT:	LA-143	1.20E+08			GRAND PARENT:	RB-92	7.83E+07
PR-144			9.89E+07		Y-93			7.73E+07
	PARENT:	CE-144	9.82E+07			PARENT:	SR-93	1.14E+08
	2ND PARENT:	PR-144M	1.38E+06		Y-94			1.23E+08
PU-238			3.40E+05			PARENT:	SR-94	1.14E+08
	2ND PARENT:	NP-238	3.98E+07		Y-95			1.28E+08
					ZR-95			1.33E+08
						PARENT:	Y-95	1.28E+08
					ZR-97			1.26E+08

Table 5.3.3-2
BVPS-1 Primary and Secondary Coolant Technical Specification Iodine and Noble Gas Concentrations

Nuclide	Primary Coolant (µCi/gm)	Secondary Coolant (µCi/gm)
I-131	7.83E-02	4.16E-02
I-132	3.08E-02	6.98E-03
I-133	1.17E-01	4.69E-02
I-134	1.71E-02	9.77E-04
I-135	6.73E-02	1.69E-02
Kr-83m	1.11E-02	
Kr-85m	3.84E-02	
Kr-85	3.37E+00	
Kr-87	2.57E-02	
Kr-88	7.18E-02	
Xe-131m	1.38E-01	
Xe-133m	1.14E-01	
Xe-133	8.42E+00	
Xe-135m	2.59E-02	
Xe-135	2.61E-01	

Table 5.3.3-3
BVPS-2 Primary and Secondary Coolant Technical Specification Iodine and Noble Gas Concentrations

Nuclide	Primary Coolant (µCi/gm)	Secondary Coolant (µCi/gm)
I-131	2.74E-01	8.33E-02
I-132	1.08E-01	1.40E-02
I-133	4.10E-01	9.39E-02
I-134	6.00E-02	1.95E-03
I-135	2.36E-01	3.39E-02
Kr-83m	3.89E-02	
Kr-85m	1.35E-01	
Kr-85	1.18E+01	
Kr-87	9.00E-02	
Kr-88	2.52E-01	
Xe-131m	4.84E-01	
Xe-133m	3.99E-01	
Xe-133	2.95E+01	
Xe-135m	9.09E-02	
Xe-135	9.16E-01	

Table 5.3.4-1 BVPS Site Boundary Atmospheric Dispersion Factors (sec/m^3)				
Exclusion Area Boundary **Averaging Period**				
Release Point	**0-2 hr**			
BVPS-1 Release Points	1.04E-3			
BVPS-2 Release Points	1.25E-3			
Low Population Zone **Averaging Period**				
Release Point	**0-8 hr**	**8-24 hr**	**1-4 day**	**4-30 day**
BVPS 1&2 Release Points	6.04E-5	4.33E-5	2.10E-5	7.44E-6

Table 5.3.4-2
BVPS-1 On-Site Atmospheric Dispersion Factors (sec/m^3)

Release	Receptor	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
U 1 Contain. Edge	BVPS-1 CR Intake	7.48E-04	5.77E-04	2.53E-04	2.00E-04	1.78E-04
U 1 Containment Top	BVPS-1 CR Intake	8.16E-04	5.78E-04	2.27E-04	1.71E-04	1.47E-04
U 1 RWST Vent	BVPS-1 CR Intake	7.34E-04	6.17E-04	2.54E-04	1.96E-04	1.57E-04
U 1 MS Relief Valves	BVPS-1 CR Intake	1.24E-03	9.94E-04	4.08E-04	3.03E-04	2.51E-04
U 1 Contain. Edge	BVPS-2 CR Intake	4.88E-04	4.07E-04	1.79E-04	1.41E-04	1.22E-04
U 1 Containment Top	BVPS-2 CR Intake	5.93E-04	4.63E-04	1.84E-04	1.34E-04	1.16E-04
U 1 RWST Vent	BVPS-2 CR Intake	4.76E-04	4.10E-04	1.70E-04	1.33E-04	1.07E-04
U 1 MS Relief Valves	BVPS-2 CR Intake	7.46E-04	6.31E-04	2.62E-04	1.98E-04	1.62E-04
U 1 Contain. Edge	BVPS-2 Aux. Bldg. NW Corner	3.34E-04	2.85E-04	1.23E-04	9.62E-05	8.37E-05
U 1 Containment Top	BVPS-2 Aux. Bldg. NW Corner	4.37E-04	3.41E-04	1.39E-04	1.02E-04	8.79E-05
U 1 RWST Vent	BVPS-2 Aux. Bldg. NW Corner	3.23E-04	2.83E-04	1.18E-04	9.32E-05	7.52E-05
U 1 Contain. Edge	BVPS-1 Service Bldg.	1.90E-03	1.57E-03	4.54E-04	5.08E-04	4.55E-04
U 1 Containment Top	BVPS-1 Service Bldg.	1.64E-03	8.59E-04	3.35E-04	2.71E-04	2.29E-04
U 1 RWST Vent	BVPS-1 Service Bldg.	2.37E-03	1.88E-03	7.58E-04	5.71E-04	4.48E-04
U 1 Contain. Edge	ERF Norm. Intake	4.53E-05	2.97E-05	1.41E-05	1.23E-05	1.09E-05
U 1 Containment Top	ERF Norm. Intake	4.57E-05	3.74E-05	1.50E-05	1.44E-05	1.23E-05
U 1 RWST Vent	ERF Norm. Intake	4.53E-05	2.87E-05	1.39E-05	1.21E-05	1.05E-05
U 1 Contain. Edge	ERF Edge Closest to Containment	4.70E-05	3.16E-05	1.54E-05	1.32E-05	1.14E-05
U 1 Containment Top	ERF Edge Closest to Containment	5.00E-05	3.94E-05	1.62E-05	1.52E-05	1.30E-05
U 1 RWST Vent	ERF Edge Closest to Containment	4.54E-05	3.14E-05	1.50E-05	1.29E-05	1.13E-05

Note: The Control Room In-leakage χ/Q values can be represented by the Control Room air intake χ/Q values. The higher values from among the BVPS-1 and BVPS-2 Control Room Intake χ/Qs are conservatively used for this purpose.

Table 5.3.4-3
BVPS-2 On-Site Atmospheric Dispersion Factors (sec/m^3)

Release	Receptor	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
U 2 Contain. Edge	BVPS-1 CR Intake	3.19E-04	2.38E-04	1.06E-04	8.08E-05	6.19E-05
U 2 Containment Top	BVPS-1 CR Intake	3.83E-04	3.10E-04	1.34E-04	9.83E-05	6.65E-05
U 2 RWST Vent	BVPS-1 CR Intake	1.70E-04	1.30E-04	5.56E-05	4.40E-05	3.31E-05
U 2 MS Relief Valves	BVPS-1 CR Intake	3.33E-04	2.38E-04	1.09E-04	7.88E-05	5.66E-05
U 2 Contain. Edge	BVPS-2 CR Intake	4.82E-04	3.59E-04	1.55E-04	1.21E-04	9.18E-05
U 2 Containment Top	BVPS-2 CR Intake	5.56E-04	4.45E-04	1.91E-04	1.39E-04	9.35E-05
U 2 RWST Vent	BVPS-2 CR Intake	2.18E-04	1.58E-04	7.31E-05	5.53E-05	4.12E-05
U 2 MS Relief Valves	BVPS-2 CR Intake	5.01E-04	3.58E-04	1.61E-04	1.19E-04	8.32E-05
U 2 Contain. Edge	BVPS-2 Aux. Bldg. NW Corner	9.12E-04	7.13E-04	3.05E-04	2.35E-04	1.79E-04
U 2 Containment Top	BVPS-2 Aux. Bldg. NW Corner	1.14E-03	8.87E-04	3.83E-04	2.74E-04	1.83E-04
U 2 RWST Vent	BVPS-2 Aux. Bldg. NW Corner	3.19E-04	2.25E-04	1.06E-04	7.95E-05	5.84E-05
U 2 Contain. Edge	BVPS-1 Service Bldg.	1.96E-04	1.54E-04	6.37E-05	5.05E-05	3.89E-05
U 2 Containment Top	BVPS-1 Service Bldg.	2.46E-04	2.07E-04	8.84E-05	6.56E-05	4.49E-05
U 2 RWST Vent	BVPS-1 Service Bldg.	1.24E-04	9.81E-05	4.10E-05	3.24E-05	2.51E-05
U 2 Contain. Edge	ERF Norm. Intake	6.02E-05	4.67E-05	2.22E-05	1.78E-05	1.59E-05
U 2 Containment Top	ERF Norm. Intake	6.16E-05	5.36E-05	2.42E-05	2.08E-05	1.81E-05
U 2 RWST Vent	ERF Norm. Intake	7.28E-05	6.58E-05	3.01E-05	2.31E-05	2.08E-05
U 2 Contain. Edge	ERF Edge Closest to Containment	6.72E-05	5.69E-05	2.65E-05	2.13E-05	1.89E-05
U 2 Containment Top	ERF Edge Closest to Containment	7.22E-05	6.43E-05	2.96E-05	2.48E-05	2.15E-05
U 2 RWST Vent	ERF Edge Closest to Containment	9.42E-05	8.37E-05	3.81E-05	2.97E-05	2.58E-05

Note: The Control Room In-leakage χ/Q values can be represented by the Control Room air intake χ/Q values. The higher values from among the BVPS-1 and BVPS-2 Control Room Intake χ/Qs are conservatively used for this purpose.

Table 5.3.6-1 Analysis Assumptions & Key Parameter Values BVPS Common Control Room LOCA & CREA Analyses	
Control Room Parameters	
Free Volume	173,000 ft^3
Unfiltered Normal Operation Intake	500 cfm
Isolation Mode Inleakage	300 cfm
Emergency Intake Rate	600 to1030 cfm
Emergency Recirculation Rate	Not Credited
Emergency Intake Filter Efficiency	99% (aerosols) 98% (elemental/organic iodine)
Emergency Recirculation Filter Efficiency	NA
Unfiltered Emergency Mode Inleakage	30 cfm
Occupancy Factors	0-24 hr (1.0) 1 - 4 d (0.6) 4-30 d (0.4)
Operator Breathing Rate	0-30 d (3.5E-04 m^3/sec)
Delay in Initiation of Control Room Emergency Ventilation Considering a LOOP	
Auto-Start on receipt of CIB (LOCA only)	
CR isolated (includes diesel start up/sequencing)	T=77 seconds
CR in emergency pressurization mode (not credited in analysis)	T=137 seconds
Manual	
CR isolated and in emergency pressurization mode (CREA)	T=30 minutes
CR isolated (includes diesel start up/sequencing)	T=77 seconds

Table 5.3.6-2 Analysis Assumptions & Key Parameter Values Loss of Coolant Accident	
Containment Leakage Parameters	
Power Level	2918 MWT
Minimum Free Volume	1,750,000 ft^3
Sprayed Fraction	63%
Spray Period	720 sec to 4 days
Mixing Rate	2 Unsprayed vol/hr
Containment Leakrate (0–24 hr)	0.1% vol fractions per day
Containment Leakrate (1-30 day)	0.05% vol fractions per day
Maximum DF for Elemental Iodine	200
Sump/Recirculation Spray pH	≥ 7.0
Fuel Activity Release Fractions	Per Reg. Guide 1.183
Fuel Release Timing (gap)	Onset: 30 sec Duration: 0.5 hr
Fuel Release Timing (Early-In-Vessel)	Onset: 0.5 hr Duration: 1.3 hr
Chemical Form of Iodine Released	4.85% elemental 95% particulate 0.15% organic
Spray Removal Constants	Figure 5.3.6-1 & 5.3.6-2
Core Activity	Table 5.3.3-1
Release Point	Containment Outer Wall / SLCRS Vent (Containment Top)

Table 5.3.6-2 (Continued) Analysis Assumptions & Key Parameter Values Loss of Coolant Accident	
ECCS/RWST Leakage Parameters	
Minimum Sump Volume (5-30 min) (0.5-2 hr) (2 hr-30 day)	9800 ft^3 (5.77E5 lbm) 28,600 ft^3 (1.71E6 lbm) 65,600 ft^3 (4.01E6 lbm)
<u>ESF Leakage</u>	
ESF Leakrate	11400 cc/hr (2×expected)
Leakage Period	5 min to 30 days
Iodine Release Fraction	0.1
Chemical Form of Iodine Released	97% elemental; 3% organic
Release Point	SLCRS Vent (Containment Top) as an unfiltered PUFF release
<u>RWST (backleakage)</u>	
Sump water back-flow to RWST	2 gpm (includes factor of 2 margin)
Onset of back-leakage	2186 seconds
Onset of RWST activity venting	5178 seconds
Iodine Release Fraction via RWST	Figure 5.3.6-3
Release Point	RWST Vent
Containment Vacuum System Parameters	
Primary Coolant Tech Spec Activity	Table 5.3.3-3
Chemical Form of Iodine Released	97% elemental; 3% organic
Containment Vacuum System Release	2200 scfm (5 sec release)
Release Point	Containment Outer Wall / SLCRS Vent (Containment Top)
CR Emergency Ventilation: Initiation Signal/Timing	
Initiation signal/Time	Assumed to be at T=0 (CIB)
Control Room Isolation	T=77 sec (Automatic)
Pressurized/Emergency Vent Mode	T=30 min (Manual initiation)

Table 5.3.6-3
Containment Thermodynamic Data – Loss of Coolant Accident

Time (sec)	Temp (^{o}F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
0.0	105.0	0.0	0.0	0.0	0.500
5.0	268.1	5.0	0.0	5.0	0.579
10.0	258.7	10.0	662517.1	10.0	1.000
15.8	267.0	15.0	896706.8	345600.0	1.000
20.0	265.2	20.0	711550.4		
30.0	260.7	30.0	421296.6		
40.0	257.5	40.0	270885.4		
50.0	255.3	50.0	194772.6		
100.0	250.0	85.0	99164.4		
300.0	238.1	140.0	88421.0		
600.0	228.6	240.0	64863.7		
1200.0	194.5	400.0	49310.9		
1800.0	175.5	700.0	66360.6		
2400.0	143.8	1200.0	41453.8		
3600.0	125.7	1700.0	41453.8		
4800.0	119.3	2200.0	14134.4		
5400.0	117.3	2700.0	7435.7		
6500.0	114.7	3200.0	5855.4		
6520.0	131.9	4700.0	5025.8		
6600.0	185.3	6200.0	4812.2		
7200.0	236.7	6700.0	103523.4		
7400.0	225.5	7200.0	97497.0		
7500.0	213.9	7700.0	43745.4		
7800.0	180.2	8200.0	14308.1		

Table 5.3.6-3 (Continued)
Containment Thermodynamic Data – Loss of Coolant Accident

Time (sec)	Temp (^{o}F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
8400.0	135.5	8700.0	6254.1		
8700.0	128.9	9200.0	5384.1		
10800.0	135.2	9700.0	5262.6		
14400.0	133.8	10200.0	2682.5		
18000.0	132.8	10700.0	4103.2		
36000.0	130.5	21200.0	4247.0		
86400.0	127.8	21900.0	4239.1		
259200.0	115.8	23400.0	4218.4		
345600.0	114.7	33900.0	4071.4		
		51900.0	3791.3		
		71400.0	3478.9		
		81900.0	3310.5		
		89400.0	3169.4		
		96900.0	3020.5		
		104400.0	2870.9		
		111900.0	2721.3		
		119400.0	2572.5		
		134400.0	2277.2		
		149400.0	1977.8		
		170400.0	1518.0		
		191400.0	1058.4		
		212400.0	596.2		
		228900.0	196.6		
		233400.0	86.6		
		242400.0	3.5		

Table 5.3.6-3 (Continued)
Containment Thermodynamic Data – Loss of Coolant Accident

Time (sec)	Temp (^{o}F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
		249900.0	2.7		
		257400.0	3.5		
		264900.0	3.5		
		281400.0	2.7		
		290400.0	0.6		
		302400.0	8.0		
		309900.0	15.1		
		321900.0	22.8		
		339900.0	34.2		
		345900.0	36.9		

Table 5.3.6-4 Analysis Assumptions & Key Parameter Values Control Rod Ejection Accident	
Containment Pathway Parameters	
Power Level	2918 MWTh
Minimum Free Volume	1.75E+6 ft^3
Containment Leakrate (0 –24 hr)	0.1% vol fractions per day
Containment Leakrate (1-30 day)	0.05% vol fractions per day
Failed Fuel Percentage	10%
Percentage of Core Inventory in Fuel Gap	10% (noble gases & halogens)
Melted Fuel Percentage	0.25%
Percentage of Core Inventory in melted fuel released to Containment Atmosphere	100% Noble Gas; 25% Halogens
Chemical Form of Iodine in Failed/Melted fuel	4.85% elemental; 95% CsI 0.15% organic
Radial Peaking Factor	1.75
Core Activity Release Timing	PUFF
Form of Failed/Melted Iodine in the Containment Atmosphere	97% elemental; 3% organic
Equilibrium Core Activity	Table 5.3.3-1
Termination of Containment Release	30 days
Environmental Release Point	Containment wall / SLCRS Vent (Containment Top)
Secondary Side Pathway Parameters	
Minimum Reactor Coolant Mass	340,711 lbm
Primary-to-Secondary Leakrate	150 gpd per SG @ STP, 450 gpd total
Termination of Primary-to-Secondary Leakage	2500 secs
Fraction of Failed/Melted Fuel	Same as Containment Pathway
Percentage of Core Inventory in melted fuel released to Reactor Coolant	100% Noble Gas; 50% Halogens

Table 5.3.6-4 (Continued) Analysis Assumptions & Key Parameter Values Control Rod Ejection Accident	
Iodine Species released to Environment	97% elemental; 3% organic
Iodine Partition Coefficient	100 (all tubes submerged)
Fraction of Noble Gas Released	1.0 (Released to Environ without holdup)
Minimum Post-Accident SG Liquid Mass	99,217 lbm per SG
Steam Releases per SG 0-150 secs:	900 lbs/sec
150-300 secs	300 lbs/sec
300-2500 secs	150 lbs/sec
2500 secs-8 hrs	776,000 lbs
Termination of Release from SGs	8 hours
Environmental Release Point	MSSVs/ADVs
CR Emergency Ventilation: Initiation Signal/Timing	
Initiation time (signal)	30 minutes by manual operation

Table 5.3.6-5 Exclusion Area Boundary and Low Population Doses (TEDE)			
Accident	**EAB Dose (rem) [1,3]**	**LPZ Dose (rem) [2]**	**SB Reg. Limit (rem)**
LOCA	14	2.5	25
Control Rod Ejection Accident[4]	3.1	1.5	6.3

Notes:

1. EAB Doses are based on the worst 2-hour period following the onset of the event.
2. LPZ Doses are based on the duration of the release.
3. The maximum 2 hr dose period for the EAB dose for the CREA and LOCA are as follows:
   - CREA: 0 to 2 hrs.
   - LOCA: 0.5 to 2.5 hr
4. Dose values are based on the containment release scenario. The dose consequences based on the secondary side release scenario are 1 Rem (EAB) and 0.1 Rem (LPZ)

Table 5.3.6-6 30 Day Integrated Control Room Doses (TEDE)		
	Control Room Operator	
Accident	**Dose (rem)**	**Reg. Limit (rem)**
LOCA[1]	2 (0.6)	5.00
Control Rod Ejection Accident[2]	1.3	5.00

Notes:

1. Portion shown in parenthesis for the LOCA represents that portion of the total dose of 2 rem that is the contribution of direct shine from contained sources/external cloud.
2. Dose values are based on the containment release scenario. The dose consequences based on the secondary side release scenario is 0.06 Rem

Table 5.3.7-1 Analysis Assumptions & Key Parameter Values BVPS Emergency Response Facility	
ERF Parameters	
Minimum Free Volume	478,610 ft^3
Filtered Normal Operation Intake	3800 cfm +/-10%
Normal Operation Filter Efficiency (HEPA)	99% (particulate)
Normal Operation Unfiltered Inleakage	2090 cfm
Emergency Recirculation Rate	3800 cfm +/-10%
Emergency Recirculation Filter Efficiency	98% (particulate) 90% (elemental/organic iodine)
Emergency Intake Rate	0 cfm
Emergency Mode Unfiltered Inleakage	910 cfm (includes 10 cfm for ingress/egress)
Occupancy Factors	0-24 hr (1.0) 1 - 4 d (0.6) 4-30 d (0.4)
Operator Breathing Rate	0-30 d (3.5E-04 m^3/sec)
Delay in Initiation of ERF Emergency Ventilation	
Manual	
ERF in emergency recirculation mode	T=30 minutes

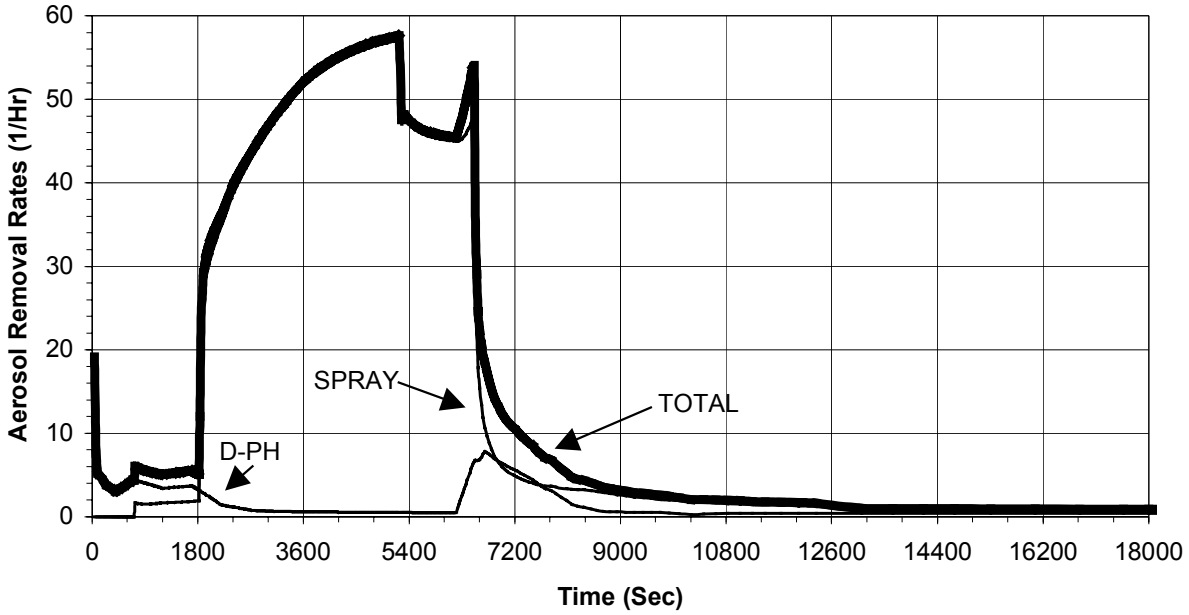

Figure 5.3.6-1
Aerosol Removal Rates Within Sprayed Region (NUREG-1465 HLDER MIN CASE)

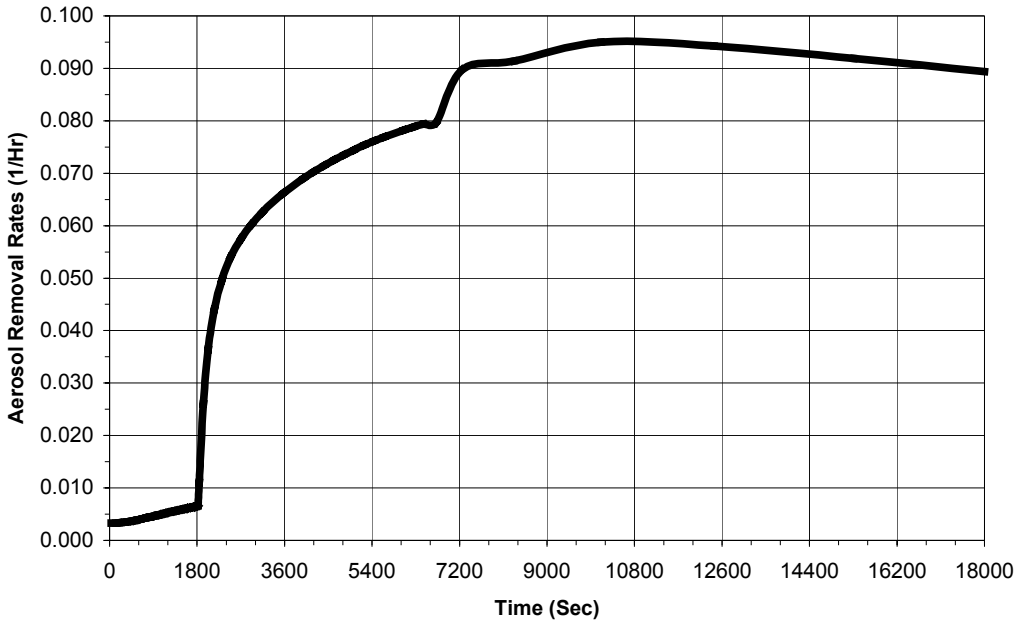

Figure 5.3.6-2
Aerosol Removal Rates Within Unsprayed Region (NUREG-1465 HLDER MIN CASE)

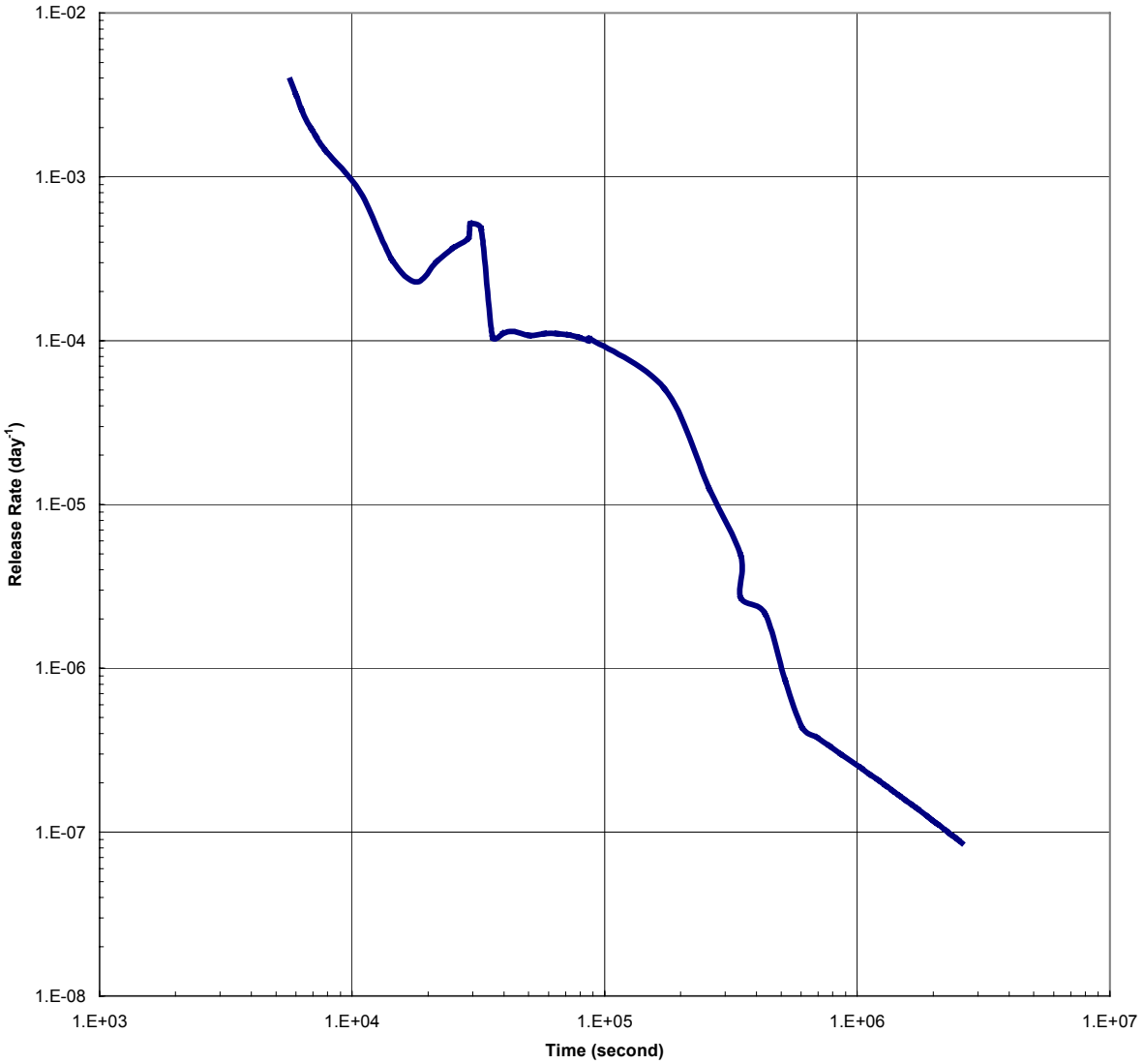

Figure 5.3.6-3
RWST Iodine Release Fractions

6.0 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 2.0 describes the analysis performed to define PCWG parameters for use as input to the LOCA and MSLB mass and energy release analysis.
- Chapter 3.0 describes the analysis performed to develop LOCA and MSLB mass and energy release data for use as input to the containment analysis.
- Chapter 4.0 describes the containment analysis performed to justify conversion of the containment pressure design basis from sub-atmospheric to atmospheric.
- Chapter 5.0 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 power uprate of BVPS-1 and BVPS-2 to 2910 MWT NSSS power. In addition, evaluations were performed for the above areas to show that the analyses performed for 2910 MWT are applicable to or bound the current power level of 2697 MWT NSSS power. These evaluations were performed to justify the potential implementation of containment conversion at the current power level (i.e., prior to implementation of power uprate) even though the containment conversion analyses described above were performed based on inputs and assumptions for the uprated power level.

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 7.0 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. Finally, Chapter 8.0 describes the environmental assessment performed to support containment conversion at the current power level. The power uprate project includes analyses and evaluations for these areas to support implementation of power uprate after or concurrent with implementation of containment conversion. These power uprate project analyses will be described in separate reports and submittals.

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 to 16 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 BVPS-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).
- Elimination of automatic control room emergency bottled air system (CREBAPS) actuation on CIB or high radiation signals.

6.1.1 LOCA Transients

The current containment designs for BVPS-1 and BVPS-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 and 16 psia
- Containment Pressure-High SAL range of 18-22 psia
- decrease in the ESFAS RWST Level-Low switchover setpoint
- minimum containment pressure 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 BVPS-2 Large Break LOCA (LBLOCA) 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 BVPS-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 BVPS-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 BVPS-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 BVPS-2. The Beaver Valley 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 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 BVPS-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 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 BVPS-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 BVPS-2.

The BVPS-1 and BVPS-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 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 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 BVPS-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 BVPS-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 BVPS-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 of 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.

For BVPS-1, three analyses are sensitive to reductions in auxiliary feedwater. These are Loss of Normal Feedwater (UFSAR Section 14.1.8), Loss of Offsite Power to the Station Auxiliaries (UFSAR Section 14.1.11) and Feedwater System Pipe Break (UFSAR Section 14.2.5.2). Each of these events has been recently analyzed at the uprated NSSS power of 2910 MWT for BVPS-1. The power uprate analyses conservatively bound current operating conditions.

The Loss of Normal Feedwater and Loss of Offsite Power to the Station Auxiliaries analyses are performed for the large uprate with the reduced auxiliary feedwater flow rates consistent with the installation of the cavitating venturis. Thus, the analysis performed for the large uprate conservatively bounds operation with the cavitating venturis installed at both current operation conditions and uprated conditions and shows acceptable results.

The feedline break relies on auxiliary feedwater as one safeguard system to prevent RCS overpressurization and hot leg boiling. The feedline break has been analyzed at the uprated power level with auxiliary feedwater flow rates consistent with the installation of the cavitating venturis both before and after isolation of auxiliary feedwater to the faulted steam generator. The reactor coolant system does not overpressurize and there is greater than 20°F of margin to boiling in the hot leg. Thus, the analysis performed for the large uprate demonstrates that operation with the cavitating venturis installed is acceptable at both current operating conditions and uprated conditions.

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 BVPS-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 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 power uprate conditions to support containment conversion and have been evaluated relative to their potential impact on Other Considerations (Chapter 6.0) and Programs (Chapter 7.0) 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 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 MOVs, 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 (as specified in 5.3.6.4) 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. The emergency control room pressurization systems used during accidents are shared by both units.

6.3.1.1 Current Plant Design, Operating Procedures, and Technical Specification

Normal Operation (Currently)

Although BVPS-1 and BVPS-2 control room areas are open to each other; in normal operation each area is served by separate HVAC systems. Thus two (2) 100% capacity HVAC systems serve the BVPS-1 area of the control room and two (2) additional 100% capacity HVAC systems serve the BVPS-2 area of the control room.

These HVAC systems maintain temperature and humidity conditions for the proper operation of control room equipment and instrumentation as well as for the comfort and efficiency of the operating personnel during both normal operation and accident conditions.

During normal operation one (1) train of HVAC serving the BVPS-1 area is in operation with the second train in standby. A similar arrangement exists for the BVPS-2 area of the control room.

Accident Events (Currently)

In the event of an accident, the control room is pressurized by emergency pressurization equipment while the above HVAC systems continue to operate but in the 100% recirculation mode. In this mode all HVAC system inlet and exhaust dampers close to avoid drawing outside air-borne radioactivity into the control room.

Initially control room pressurization is achieved by a bank of bottles containing pressurized breathing air. The bottles discharge into the control room and are capable of pressurizing the control room for 60 minutes, assuming a single bottle bank fails to discharge. One common bottle system services both BVPS-1 and BVPS-2. The bottled air system is referred to as the Control Room Bottled Air Pressurization System (CREBAPS).

After an initial 60-minute period, control room pressurization is maintained indefinitely by emergency pressurization fans and filters. The fans move outside air through prefilters, charcoal filters and particle filters before discharging it into and pressurizing the control room. The emergency pressurizing fans and filters are referred to as the Control Room Emergency Ventilation Systems (CREVS). There are three (3) 100% CREVS systems.

Current Design and Operating Procedures

The emergency control room pressurization systems are designed as follows:

A CIB signal from either unit, or High control room radiation, or manual CREBAPS actuation automatically starts the following sequence:

- The inlet and exhaust dampers on all normal HVAC systems are closed and
- CREBAPS is actuated, and
- Timers are started which send a start signal to all 3 CREVS systems after 60 minutes

CREBAPS consists of five banks of two bottles each. Each bank of two bottles contains a 60-minute supply of air equal to 25 % of the required supply to pressurize the room. Thus CREBAPS achieves single failure capability because any four of the five banks can pressurize the Control Room.

Three (3) capacity CREVS systems serve the common control room. Two (2) CREVS systems are powered from train A and train B of BVPS-2 respectively. The third CREVS system is powered from either train A or train B of BVPS-1. All three (3) CREVS systems receive automatic start signals 60 minutes after CREBAPS is activated. Any one (1) of the three CREVS systems is capable of pressurizing the entire control room.

The two (2) CREVS systems powered from BVPS-2 are fully automatic meaning that both the fans and the dampers operate automatically on the start signal. Thus either of these CREVS systems can pressurize the control room with no operator actions.

The CREVS system powered from BVPS-1 is not fully automatic. Its fan control switches are not maintained in the auto start position due to normally closed manual fan inlet and discharge dampers.

Single failure is addressed in the BVPS-2 Technical Specifications by requiring both fully automatic CREVS to be operable.

The BVPS-1 Technical Specifications requires any two (2) of the three (3) CREVS Systems to be operable. If the fully automatic CREVS should fail to start the operator can start the third system by manually opening the fan inlet and discharge damper and manually repositioning the fan start switch.

If either of the fully automatic CREVS systems is INOPERABLE, the third CREVS system provides single failure protection. In this case if the one remaining fully automatic CREVS system does not start (single failure) the operator can start the third system by manual opening of these dampers and manually repositioning the fan "pull to lock" switch. Thus in this unusual situation, three (3) distinct manual operator actions are required to activate CREVS and pressurize the control room. The manual inlet and discharge dampers located on the floor below the main control room must be opened and the control room operator must throw the fan start switch.

Current Technical Specifications

BVPS-1 Technical Specification 3 / 4.7.7 "Control Room Habitability Systems" requires the following for Modes 1, 2, 3, 4 and during movement of irradiated fuel or movement over irradiated fuel:

- Any two (2) of three (3) CREVS systems OPERABLE and
- Five (5) Banks of CREBAPS bottles OPERABLE and
- All HVAC inlet and exhaust dampers OPERABLE and capable of automatic closure on a CIB or Control Room High radiation isolation signal or secured in a closed position with power removed

  A 7 day LCO action if CREBAPS or HVAC isolation dampers are no longer single failure proof or any 2 CREVS systems are INOPERABLE

  A 3.0.3 action if CREBAPS or HVAC isolation dampers are not capable of their design function or all three CREVS are INOPERABLE.

BVPS-2 Technical Specification 3 / 4.7.7 "Control Room Emergency Air Cleanup and Pressurization System" requires the following for Modes 1,2,3,4 and during movement of recently irradiated fuel or movement over recently irradiated fuel:

- Two (2) automatic CREVS systems OPERABLE and

- Five (5) Banks of CREBAPS bottles OPERABLE and

- All HVAC inlet and exhaust dampers OPERABLE by automatic actuation on a CIB signal or secured in a closed position with power removed

  A 7 day LCO action if CREBAPS or HVAC isolation dampers are no longer single failure proof or any one (1) of the automatic CREVS systems is INOPERABLE

  A 3.0.3 action if CREBAPS or HVAC isolation dampers are not capable of their design function or the two (2) automatic CREVS are INOPERABLE

6.3.1.2 Proposed Changes

With implementation of atmospheric containment and the use of alternative source term, CREBAPS will no longer be required to provide pressurization.

The requirement for CREBAPS is being removed from the Technical Specifications and the control room habitability will be achieved as follows:

- Normal HVAC systems will isolate and go on 100% recirculation on a CIB actuation, high Control Room Radiation, or manual operator action, as in the current design.

- CREVS 60 minute auto start delay will be eliminated.

- A CREVS pressurization fan will start on CIB, Control Room Radiation, or Manual activation.

- Technical Specifications for BVPS-1 CREVS filter efficiencies are changed to equal those for BVPS-2.

- If one fully automatic, CREVS train is inoperable and a single failure disables the second fully automatic CREVS train, then manual operator action will be required to start the third CREVS before 30 minutes.

6.3.2 Supplementary Leak Collection and Release System

The BVPS-1 Supplemental Leak Collection and Release System (SLCRS) is composed of two leak collection exhaust fans. One of the two leak collection exhaust fans is normally used to exhaust various areas with the filter bank bypassed. The other fan and filter bank is used as a standby.

The primary function of the BVPS-1 system is to ensure that radioactive leakage from the primary containment following a DBA, or radioactive release due to a fuel handling accident, or

radioactive material released in the waste gas storage area is collected and filtered for iodine removal prior to discharge to the atmosphere. The SLCRS collects potential radioactive contamination from the following areas: Charging pump cubicles, fuel building, areas contiguous to the containment, including the hydrogen recombiner room, MCC area, personnel hatch area, north and west safeguards, waste gas storage and the pipe tunnel. An important secondary function of the BVPS-1 system is the removal of heat from areas contiguous to the containment where equipment important to safety is located (i.e., the charging pump cubicles, the low head safety injection pump cubicles, the recirculation spray pump cubicles, the auxiliary feed water pump room, the east and west cable vaults, the MCC room and the safeguards pipe tunnel). Following a loss of offsite power, the SLCRS is the only available means of assuring that components in these areas do not exceed design temperatures.

During normal operation the exhaust does not flow through filters. On a CIA signal or a high radiation signal from monitors in the ventilation exhaust from the fuel building, the waste gas storage area or from areas contiguous to the containment with the exception of the main steam valve cubicle, the leak collection system is diverted to the filter banks. The system is designated as safety related.

The BVPS-2 Supplementary Leak Collection System is composed of the Leak Collection Normal Exhaust System and the Leak Collection Filtered Exhaust System.

The function of the BVPS-2 Supplementary Leak Collection System is to collect containment leakage to the cable vault and rod control area, the charging pump cubicle, component cooling water pump area, safeguards area, auxiliary building, pipe tunnel, equipment hatch, personnel access hatch area solid waste handling building and the fuel building. An important secondary function of the SLCRS is the removal of heat from the areas contiguous to the containment where equipment important to safety is located (i.e., the charging pump cubicles, cable vault and rod control area and the component cooling pump area). Following the loss of offsite power, the SLCRS is the only available means of assuring that components in these areas do not exceed design conditions.

During normal operation, the exhaust to the Leak Collection Normal Exhaust System does not flow through filters. On a CIA signal or a high-high radiation signal from monitors located in the ventilation exhausts from areas contiguous to the containment, the air that is normally exhausted by the leak collection normal exhaust fans is diverted to the Leak Collection Filtered Exhaust System. The Leak Collection Normal Exhaust System is non-safety related and the Leak Collection Filtered Exhaust System is designated as nuclear safety-related.

The capacity of the BVPS-1 and BVPS-2 SLCRS exceeds the air in-leakage to the areas being exhausted. The excess capacity ensures a negative pressure will be maintained in areas contiguous to the containment wall, and therefore the environmental condition in those areas will likewise be maintained. SLCRS was evaluated for power uprate, atmospheric containment conversion without taking credit for the installed HEPA or charcoal filter absorbers.

In addition, 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 doses due to increased filter shine. EQ zones and vital access routes in the vicinity of the SLCRS filters were reviewed and found to remain within allowable limits.

6.3.3 Containment Ventilation System

The containment atmosphere recirculation system is designed to maintain the bulk air temperature in the containment suitable for equipment operation during normal plant operation. The system is non-safety related.

The containment atmosphere recirculation 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. Each fan is provided with two speeds, one for normal operation and the other for containment pressure test condition operation.

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 containment atmosphere recirculation system is capable of operating under the pressure range conditions, temperature conditions, and at the increased heat loads associated with the core power uprate 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 (16 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 BVPS-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 on 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 containments 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 no longer be required to bring the containment atmosphere to a sub-atmospheric condition to meet analysis requirements. Thus the remaining purpose of the steam jet air ejectors would be to bring the containment to a sub-atmospheric condition to test the containment air in-leakage rate if desired. Changes to the current operating procedures for start up will be required.

Currently, the containment vacuum pumps establish and maintain the reactor containment internal pressure at sub-atmospheric conditions during normal operations. At this sub-atmospheric condition, the containment operates within current analysis requirements. After the conversion to an atmospheric containment, the containment vacuum pumps may be required to maintain the containment internal pressure at the desired operating value within revised analysis requirements. 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 require revision for the revised operating range.

The containment vacuum pumps were originally sized to maintain containment internal pressure based on an expected load (e.g., containment in-leakage). The conversion to an atmospheric containment would be expected to reduce the in-leakage. Thus the vacuum pumps capacity is expected to remain adequate after the containment conversion.

The Containment Vacuum Systems have been reviewed for impact on design, operation, technical specifications, UFSAR, and design basis documents as a result of converting to an atmospheric pressure containment. A modification to this system will be required as a result of this change which would be changing the narrow range containment internal pressure range (BVPS-1). The BVPS-2 narrow range remains adequate.

The safety function of the systems is that the containment isolation valves are currently designed to close on a containment isolation signal and that will not change. Therefore, no change will be required to safety-related portions of the systems after the conversion to atmospheric containments.

6.3.7 Containment Instrument Air System

The BVPS-1 Containment Instrument Air System is comprised of redundant air compressors (including unloaders), redundant water after coolers, redundant compressor receivers, a system receiver, two moisture removing prefilters, and parallel air dryers. The above BVPS-1 components are located inside containment. The BVPS-1 is normally operated using the above in containment compressors. Since these compressors take suction directly from the containment atmosphere, they make no net change in the containment air volume and will not impact containment pressure. These components are non-safety related. These compressors were designed for service at atmospheric conditions and will thus perform equally well after conversion to atmospheric conditions. Setpoints have been evaluated and found to be acceptable without change.

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.

There will be no changes required for operation with an atmospheric containment, no anticipated increase in required maintenance and no changes required for operation procedures. Thus, conversion to atmospheric containment will not significantly impact the operation of these compressors. Setpoints have been evaluated and found acceptable without change.

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 360 degree 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 deletion of a cutback system at BVPS-1, a modest increase in the maximum spray temperature, and a higher CIB setpoint. The purpose of the cutback system was to reduce QS flow on a low level setpoint on the RWST to extend the spray duration for sub-atmospheric pressure control. This is no longer required and 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 power and uprated power, 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. In the event that a LOCA does not achieve the CIB set pressure manual operator action may be required to actuate the containment spray system. Ample time exists for operator action and no changes are required to the operating procedures as described in the UFSAR. 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 header feeds a 180 degree spray ring header located beneath the top of the reactor containment. 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. Elimination of the requirement to maintain sub-atmospheric conditions following a LOCA allows for a reduction in the heat removal requirements of the system. This is limited to some degree by NPSH considerations and sump temperature effects on piping system analyses. However, some of this margin is utilized to allow for a small reduction in the required service water flow to the RS heat exchangers. The system has been analyzed for the revised operating conditions (including atmospheric conversion and uprated power) and successful performance of the design basis is demonstrated.

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 [ref. NuReg/CR-5732 and 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 [ref. 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 is exhausted. The Chemical Addition System is described in UFSAR-1 Section 6.4.2, and in UFSAR-2 Section 6.2.2.

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.

The changes associated with atmospheric conversion have been evaluated for their effect on the performance of the Chemical Addition System, specifically changes to the automatic SI switchover to recirculation setpoints on RWST level, and deletion of the Quench Spray cutback at BVPS-1. Evaluations of the long-term sump pH also considered the effect of acids created during the LOCA event by radiolysis of the sump water and irradiation of chlorinated biphenyl products used for fire protection in cable jackets inside containment. The amount of exposed cable jacketing inside containment was conservatively estimated based on the total combustible loading due to cable insulation inventory in the Beaver Valley Fire Hazards Analysis. The results demonstrate that the required pH range will be met for at least 30 days following a LOCA.

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 following phases; 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 elsewhere in 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 4.0 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.

6.3.12 Piping Systems

The piping systems (pipe stress, pipe supports, and equipment nozzle and supports) evaluated for BVPS-1 and BVPS-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 4.1.7.8 and Table 4.1.20.

The evaluations for recirculation spray and 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 revised containment analysis for the containment atmospheric conversion. The piping downstream of the recirculation heat exchanger, was evaluated for effects associated with hotter water. The evaluations performed have concluded that the river/service water piping systems, needs no modifications. Thus, remains acceptable and will continue to satisfy design basis requirements in accordance with applicable 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 during normal plant operation. The third pump is an installed spare. One of three pumps is required for the normal operation or 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. The containment analysis described in Chapter 4 has demonstrated that the minimum flow requirement can be reduced from 8,000 gpm to 7,800 gpm. A corresponding change is being made in Technical Specification 3.6.2.2. 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 out of the three pumps are required for normal operation and one pump is required for safe shutdown. The third pump is an installed spare. 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. The containment analysis described in Chapter 4 has demonstrated that the minimum flow requirement can be reduced from 11,000 gpm to 10,600 gpm. A corresponding change is being made in Technical Specification 3.6.2.2. 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 Power Uprate 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 BVPS-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 Wide Range 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 safety-related Wide Range 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.

Wide Range 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 also used in the Emergency Operating Procedures.

For BVPS-1 only, 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 BVPS-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, revised radiological 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 1A-300/2A-172.

7.0 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 Beaver Valley Unit 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. Beaver Valley Power Station 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 Unit 1 (BVPS-1) utilizes on-site Emergency AC (EAC) Power from Beaver Valley Power Station – Unit 2 (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 Beaver Valley Power Station – (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 BVPS-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 is not being changed. 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 BVPS-2 motor operated valves. The containment conversion and power uprate 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 atmospheric containment conversion.

- The calculations also conservatively postulate a piping break so the downstream side of a valve would be subjected to atmospheric pressure. Thus, for valves with downstream piping inside containment, a sub-atmospheric containment results in a increase in the calculated valve DP.

Therefore, the MOV DP calculations performed for a sub-atmospheric containment remain conservative (yields higher valve DP) for the atmospheric containment.

Atmospheric containment 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 the loss of motor starting torque at elevated temperatures. The existing evaluations were found to remain adequate. This comparison indicates the original evaluations remain adequately conservative.

Therefore, the elimination of the sub-atmospheric containment will not adversely affect any MOV currently within the 89-10 program. The atmospheric 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.

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 BVPS-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 License Amendment Requests 300 (BVPS-1) and 172 (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 = 42.5 psig and BVPS-2 P_a = 43.4 psig.

Since the BVPS-2 re-calculated P_a is lower, the current leakage testing bounds the new P_a. 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 of 42.6 psig, which bounds the new BVPS-1 P_a. Thus, additional/confirmatory Type A testing is not required.

BVPS-1 Type B testing of electrical penetrations is performed at a test pressure of 45 psig, 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 of 42.0 psig. Since this testing utilizes 0.1% accuracy 0-100 psig test pressure gauges, the minimum actual test pressure is 41.9 psig. The expected increase in the local leak rate test results as a result of the higher P_a is estimated to be $[(42.5 + 14.7) / (41.9 + 14.7)]^{1/2}$, or approximately 0.5%.

Technical Specifications require that the combined leakage rate for all penetrations subject to Type B and Type C tests to be less than 0.60 the maximum allowable containment leakage rate (L_a) at the calculated peak containment DBA internal pressure (P_a). The current combined leakage rate is approximately 27% of the acceptance criteria. Thus, a 0.5% increase in the current combined leakage rate is not significant.

Since the containment leakage test frequencies are based upon the 10 CFR 50, Appendix J, Option B Performance-Based Requirements, not all penetrations require Type B and Type C tested every refueling outage. 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, during the refueling outage prior to implementation of the atmospheric containment conversion.

Air locks are typically tested by pressurizing the interior of the airlock barrel and determining the overall airlock leakage. The emergency escape hatch containment side door design includes eight test clamps, which are rotated to the test position to hold the door against the internal (P_a) pressure. The emergency air lock door seals are also tested at a lower (10.0 psig) nominal pressure. This lower pressure corresponds to the maximum allowable test pressure without rotating the eight test clamps to the test position, and, therefore, is not affected by the atmospheric containment conversion.

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 seals have 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.

- The discharge pressure of the hydrogen recombiner is now slightly above atmospheric pressure but still remains well below the design capability of the piping. Periodic testing will continue to verify the integrity of the piping and containment isolation valves.

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 Beaver Valley Unit 1 & 2 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 sub-atmospheric pressure in the Containment to operating with atmospheric pressure (12.0 to 16.0 psia) in the Containment.

The environmental qualification of the Beaver Valley Power Station Unit 1 & 2 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 conversion to an atmospheric 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 Beaver Valley Unit 1 & 2. 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 system operates under the long-term (> 24-hour) post-accident temperature. Although the revised analysis shows that MSLB events within Containment are enveloped by the existing qualification temperature profile a slight change is necessary to the temperature profile within the Containment during and following the LOCA transient.

The containment accident qualification temperature and the operating time during the transient (e.g., peak temperature and first 24-hour conditions) and the long-term post-accident conditions have been determined to be acceptable. The qualification of the equipment for these two accident phases is addressed using different approaches.

The revised accident analysis provides temperature profiles for different containment areas (i.e., volumes defined by MAAP5 analysis nodes. The worst case LOCA containment temperature transients are enveloped in every area by the existing equipment qualification temperature transient with the following exceptions. These exceptions are the incore instrumentation room (Node 3) and the reactor coolant loop C cubicle (Node 5 – the break node). The equipment that is located in these nodes are demonstrated to be qualified under the new temperature conditions by direct comparison of each specific equipment's qualification report accident test temperature profile to the corresponding MAAP5 temperature profile for its node.

The longer-term temperatures determined by the MAAP5 analysis exceeded the temperature used for the post accident operating time (PAOT) qualification of the equipment in the same two nodes. These nodes have a slight increase over the qualification temperature existing PAOT qualification.

The equipment that is specifically located in these nodes and the equipment that is qualified for all locations in the containment (e.g., cables, splices) have been determined to have an acceptable PAOT exceeding 180 days (or its required PAOT). The use of 180 days is consistent with the current licensing basis for BVPS. This evaluation was performed using the Arrhenius equation to convert the equipment's accident test temperature and time to an equivalent time at the new PAOT temperature. This evaluation used the equipment qualification report accident test

temperatures. This equivalency conversion is comparable to the methodology utilized in evaluations performed for initial 10CFR50.49 compliance, and is consistent with the industry approach described in Chapter 4 of TR-100516, "Nuclear Power Plant Equipment Qualification Reference Manual."

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 5.3.7.2.

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-1 and BVPS-2 is based on a sub-atmospheric containment design. An engineering 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 engineering evaluation is limited to the impact of the containment conversion on the contributors to LERF identified in the PRA. The PRA of record used in this engineering assessment was the BVPS-1 PRA Update Report, Issue 1, dated June 30, 1998 to perform an evaluation on both BVPS-1 and BVPS-2.

The PRA model identifies the following contributors to LERF:

Contributor	% of Total LERF	Total LERF
Interfacing System LOCA (containment bypass):	44.01%	
In-Vessel Steam Explosions:	41.03%	
Early Hydrogen Burns:	9.28%	
Induced SG Tube Ruptures (containment bypass):	5.16%	
Direct Containment Heating:	0.52%	
Total	100%	9.14E-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 allowed 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. With the conversion to an atmospheric containment, this “operational” diagnostic of containment isolation failures can no longer be credited.

Containment Baseline Pressure and Temperature – The sub-atmospheric design provided a pre-accident containment pressure and temperature baseline that was below that which would be present with an 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 from a sub-atmospheric design 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 LERF contribution from in-vessel steam explosions is based on damage to the containment liner as a result of the reactor vessel upper head being accelerated by an explosive interaction of steam and molten core material inside the reactor vessel. The containment conditions (atmospheric vs. sub-atmospheric) play no role in determining the contribution to LERF from this failure mode.

Therefore, only the LERF contributions from early hydrogen burns, direct containment heating and containment isolation failures need to be explicitly considered.

The LERF contribution from early hydrogen burns is controlled by sequences in which there is a rapid release of a large amount of hydrogen from the reactor coolant system to the containment such that the containment hydrogen concentration exceeds 12 volume percent. In these cases, containment failure is assumed to be caused by an explosive detonation of the hydrogen that results in excessive impact loadings on the containment. Since the current PRA does not assume an oxygen limitation in the assessment and since the baseline containment pressure is unimportant in assessing impact loadings, the containment conversion would not impact this LERF contributor.

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. From the information presented in the PRA, an increase in the baseline containment pressure of 3.5 psia (from the tech spec maximum allowable pressure for sub-atmospheric containments of 12 psia to a tech spec maximum allowable for atmospheric containments of 1 psig) would conservatively result in an increase in containment rupture failure probability by less than 50%. Since direct containment heating only contributes 0.52% to the total LERF, this would translate to a 0.26% increase in LERF.

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 8.5E-05 per reactor-year. Therefore, the increase in LERF due to pre-existing containment isolation failures is 1.5E-08 or 1.6%. 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 2%. The overall LERF from the BVPS is on the order of 1E-06 per year; therefore a 2% increase would be on the order of 2E-08 per year. This is well within the criteria established in Reg. Guide 1.174 for acceptably small risk increases.

7.13 Human Factors

The accident at Three Mile Island led to the issuance of NUREG-0700 "Guidelines for Control Room Design Reviews," and NUREG-0737, "Clarification of TMI Action Plan Requirements." Beaver Valley 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 the Station's 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 BVPS-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 BVPS-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. BV-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 Beaver Valley 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.

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	NO*
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

* Program will not change. Any changes to hardware for atmospheric containment will be included in the program as appropriate.

No Programs not impacted by containment conversion or are bounded by existing analysis.

Yes Programs impacted and changes required to address containment conversion.

8.0 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 BVPS-2, respectively.

The FES documents (References 2 and 5) are considered historical licensing documents that will not be revised. However, the bases for the conclusion in the FES are maintained through adhering to the operating license and the NPDES permit.

The BVPS-1 and BVPS-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 BVPS-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 BVPS-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.

<u>Environmental Review for Normal Operation</u>

The Environmental Report (ER-OL) and the FES present the normal operation radiological releases and resultant doses, and they provide the bases 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 BVPS-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 UFSAR 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 Section 5.0, 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.

Protection of the environment and public is assured by compliance with 10CFR50.67 and 10CFR100.

References

1. NPDES Discharge Permit No. PA0025615, Commonwealth of Pennsylvania, Department of Environmental Protection, dated September 29, 1995.

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 BVPS-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.